Chapter 11

Seismic, Fire, and Flood Risk Analyses

11.1 INTRODUCTION

Chapter 10 described the overall procedure for estimating radiological risks from external events. The objective of the present chapter is to illustrate the application of this procedure to three specific external events: earthquakes, fires, and floods. As mentioned in Chapter 10, some modifications of the procedure may be necessary, depending on the external event under study.

The external event analyses discussed in this chapter illustrate different aspects of the overall procedure. The section on seismic risk analysis (Section 11.2) emphasizes the development of hazard and fragility models for predicting the occurrence frequencies of large earthquakes (i.e., earthquakes well beyond the plant design basis) and estimating the failure frequencies of components subjected to such earthquakes. The section on fire-risk analysis (Section 11.3) presents techniques used in screening for critical hazard locations and explains the details of a fire-propagation analysis. Section 11.4, which covers flood-risk analysis, highlights the uncertainties of a hazard analysis based on sparse or questionable data and describes the techniques of hazard-source screening and fragility development.

The external events discussed in this chapter as illustrations of the overall procedure have also some historical significance. For various reasons, each of them has been studied in the past, although not to the same extent. The procedures described in Chapter 10 are directly applicable to these events. For other external events, either the analytical methods have not been developed and applied or the experience in treating the events has been highly limited. The probabilistic analyses of external events discussed in plant safety analysis reports are generally restricted to one or a few stages of the overall procedure (e.g., a hazard analysis and the evaluation of structural failure frequencies) and are aimed only at calculating the frequencies of unacceptable damage as defined by NRC regulatory documents. A complete PRA has not been the objective of the studies that are reported in the safety analysis reports.

In keeping with the spirit of this guide, this chapter reflects the current state of the art in the treatment of external events in a PRA study. No new methods or improvements are suggested.

11.2 SEISMIC RISK ANALYSIS

11.2.1 INTRODUCTION

This section describes procedures for estimating radiological risks from seismic events. Its objective is to illustrate the application of the general risk-analysis procedure for external events and to highlight the similarities and differences in the analyses of seismic risk and the risk from other external events.

The analysis of seismic risk has been receiving increased attention in recent years. It is recognized that seismic excitation has the potential for simultaneously damaging several redundant components in a nuclear power plant. The basis for the conclusion in the Reactor Safety Study (USNRC, 1975) that earthquakes are not major contributors to risk has been questioned by several experts in the fields of seismology, earthquake engineering, and probabilistic risk analysis. Seismic risk studies performed since the Reactor Safety Study have indicated that the contribution of seismic risk to the overall plant risk may not be insignificant.

Following the general procedure for a probabilistic assessment of external events, the elements of a seismic risk analysis can be identified as analyses of (1) the seismic hazard at the site, (2) the responses of plant systems and structures, (3) component fragilities, (4) plant systems and accident sequences, and (5) consequences. The results of this analysis will be used as input in defining initiating events, in developing system event trees and fault trees, in quantifying the accident sequences, and in modifying the containment event trees and consequence models to reflect the unique features of seismic events. However, in the seismic risk studies done to date, the analysts have kept the seismic risk analysis separate from the analysis of internal events in the plant-system and accident-sequence analysis. The frequencies of the release categories attributable to seismic events are combined with those stemming from internal events, and a consequence analysis is performed to calculate the total plant risk.

The evaluation of seismic risk requires information on the seismologic and geologic characteristics of the region, the capacities of structures and equipment to withstand earthquakes beyond the design bases, and the interactions between the failures of various components and systems of a nuclear power plant. Empirical data available on these aspects are limited; the use of sophisticated analytical tools to calculate the real inelastic capacities of equipment and structures is expensive and has to be done on a selective basis commensurate with the uncertainties of the overall seismic risk problem. Therefore, the procedures described in this chapter call for engineering judgment based on expert opinion to supplement sparse data and limited analyses.

The output of the seismic risk analysis will depend on the stage at which the seismic event analysis is merged with the analyses of other external and internal events. If the seismic analysis is combined with the analyses of other events at the stage of accident-sequence definition and system modeling, the output will be an estimate of the seismic hazard at the site; component fragilities; initiating events; and the information needed to modify system event and fault trees, containment-failure analysis, and the frequencies of accident sequences. If the seismic analysis is combined with the analyses of other events at the consequence-analysis stage, an initial output of the seismic risk analysis could be a curve showing the probability distribution of the annual frequency of a seismically induced core melt (Figure 11-1). If this core-melt frequency is sufficiently high, further computation of the release frequencies is warranted. In that case, the final output of the seismic risk analysis is a family of probability density functions for the annual frequency of various release categories (Figure 11-2). A result of this type forms an input to the consequence analysis described in Chapter 9. Other useful outputs of the seismic risk analysis are failure frequencies for structures, systems, and equipment as well as the accident sequences that dominate the seismic risk. These permit the identification of the major contributors to core-melt and release-category frequencies.

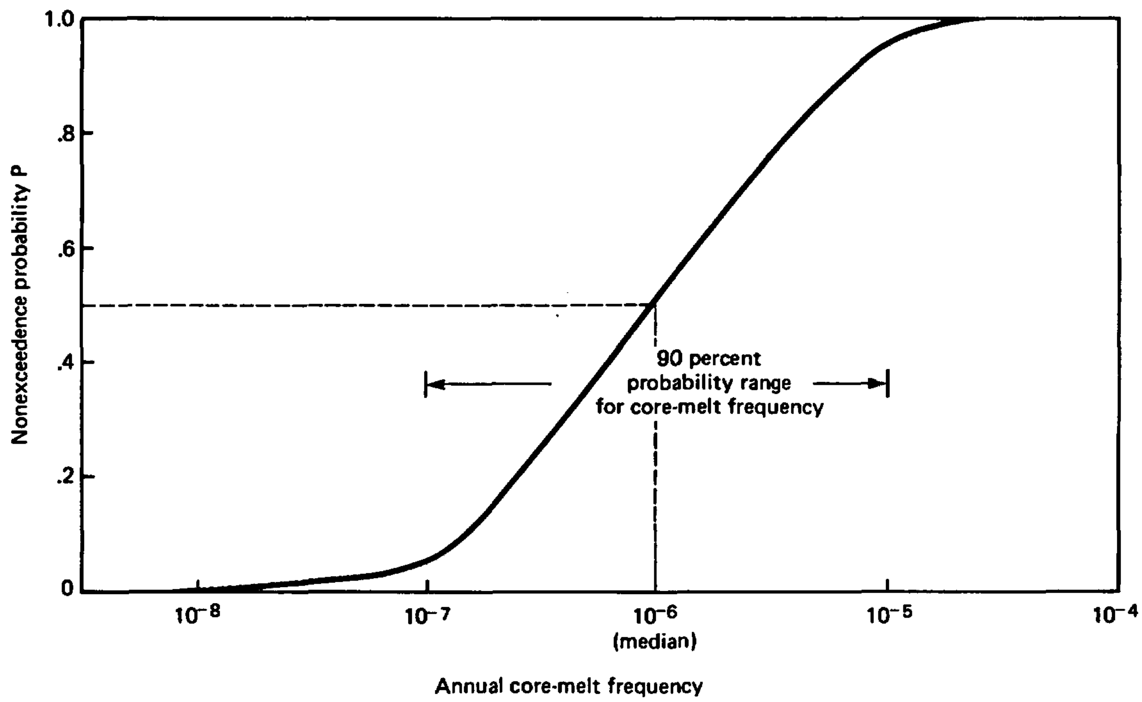

Figure 11-1. Probability distribution for the annual frequency of a seismically induced core melt in a hypothetical nuclear power plant.

11.2.2 HISTORICAL BACKGROUND

Several studies of seismic risk have been performed for nuclear power plants. The Reactor Safety Study (USNRC, 1975) examined a generic safety system consisting of two components in parallel. It used a single fragility curve based on the work of Newmark (1975) along with the seismic hazard estimates developed by Hsieh et al. (1975) for a site in the Eastern United

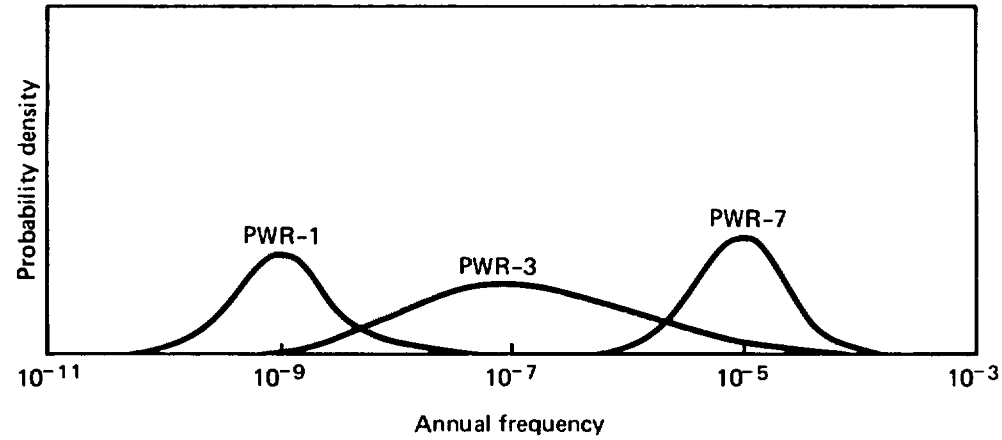

Figure 11-2. Probability density functions for release frequencies from seismic events for three release categories: PWR-1, PWR-3, and PWR-7.

States. The possibility of common-mode failures was admitted by assuming that the system-failure frequency for any given earthquake ground acceleration is not the square of the component-failure frequency (as it would be if the failures of components were independent events) but rather the component-failure frequency raised to the 1.5th power (i.e., the geometric mean of the failure frequencies of two components). The frequency of an earthquake-induced core melt in reactors designed for a safe-shutdown-earthquake (SSE) ground acceleration of 0.20g was estimated to be in the range of 1×10^{-8} to 1×10^{-6} per year. It was concluded that, in comparison with other reactor accidents, seismic events are not significant contributors to risk. This conclusion has since been seriously questioned on the grounds that in the Reactor Safety Study nuclear safety systems were modeled in a simplistic fashion, seismic safety margins were incorrectly calculated, and common-cause failures were given an approximate treatment.

A seismic risk study conducted by Anderson et al. (1975) for Canadian nuclear power plants included a generic safety system consisting of three independent components in parallel. Four fragility curves were developed to reflect differences in design practices and assumptions. The basic component-fragility curve was simply a straight line passing through the failure frequencies of 0.1 and 0.9 at response levels of 1.0 and 4.0 times the design response. Three other variations were included: the "weak" component, the "strong" component, and the "brittle" component. The frequency of system failure under earthquake conditions was calculated. The seismic contribution to risk was found to be important.

In their seismic risk study, Hsieh and Okrent (1976) considered a generic system of 10 components in series. The components were categorized into three groups of two, three, and five, on the basis of differences in the fragility attributed to degradation. Furthermore, two classes of design errors--minimal and maximal--were considered. It was shown that the potential for design error would greatly influence the frequency of system failure.

Described below are the seismic risk studies performed for the Diablo Canyon, Oyster Creek, and Big Rock Point nuclear power plants. Other generic and plant-specific seismic risk studies have been reported (Cornell and Newmark, 1978; Clinch River Breeder Reactor Plant, 1977). More-recent and ongoing studies are discussed in Section 11.2.11.

11.2.2.1 Diablo Canyon Seismic Risk Study

The first plant- and site-specific seismic risk study* was carried out in 1977 by the Pacific Gas & Electric Company for the Diablo Canyon plant (see also Ang and Newmark, 1977). The study was prompted by the realization that the Hosgri Fault could be a major seismic threat to the plant. The following five initiating events were postulated for each earthquake:

1. Transients that require a successful cooldown for the reactor-coolant system (RCS).

2. A small-small LOCA (RCS pipe break of 1/2 to 2 inches).

3. A small LOCA (RCS pipe break of 2 to 6 inches).

4. A large LOCA (RCS pipe break of more than 6 inches).

5. Reactor-vessel rupture.

The frequency of occurrence for each initiating event was determined by considering the fragilities of various components whose failures constitute the initiating event. For each initiating event an event tree was developed and fed into a containment event tree. The frequency of occurrence for each accident sequence was calculated from the failure frequencies of the safety systems. Detailed fault trees were constructed for a number of systems: containment-spray injection, emergency core-cooling injection, containment-spray recirculation and fan coolers, containment heat removal, emergency core-cooling recirculation, auxiliary feedwater, high-pressure injection, and electric power.

The plant structures and piping at Diablo Canyon were analyzed for the design earthquake, an earthquake that is double the design earthquake, and the postulated Hosgri Fault earthquake. The peak ground accelerations for these earthquakes were taken to be 0.2g, 0.4g, and 0.75g, respectively. Assuming the seismic stress to be a linear function of ground acceleration, the ground acceleration at which the piping would reach the code-allowable stress and the ultimate strength of the material were calculated, as was the ground acceleration at which structures would reach the first yield stress

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*Although this study was not part of a probabilistic risk assessment for the plant, it has the historical significance of being the first detailed seismic risk study.

and the ultimate strength. This was done for various locations in the structures and for various segments of the piping. For mechanical and electrical equipment, the seismic qualification acceleration levels were related to the ground-acceleration levels by assuming that the acceleration of the floor on which the equipment was mounted was a linear function of elevation. It may be noted that this assumption is a gross approximation for nuclear plant structures.

Component fragilities were expressed in terms of the effective peak ground acceleration. Two basic forms of fragility curves were used: a ramp-function curve and a step-function curve. For component fragilities described by a ramp function, zero frequency of failure was assumed below a specific acceleration level a, corresponding to the first yield stress in the structure, or the code-allowable stress in the piping, or to the seismic qualification level of the mechanical and electrical equipment; above a specific acceleration level b, corresponding to the ultimate strength of the structure and piping, the failure frequency was unity. For ground accelerations between a and b, the failure frequency was varied linearly between zero and unity. For component fragilities described by a step function, the frequency of failure below the acceleration value a was taken to be zero; above the value of a, it was taken to be unity. The ramp function was generally used for components, such as piping and structures, qualified by stress analysis; the step function was generally used for components, such as mechanical and electrical equipment, qualified by testing.

Variations in component quality and design and fabrication errors were accounted for by including nonzero frequencies for effective peak-ground-acceleration values smaller than a. Typical values of a and b for the containment structure were 0.9g and 1.5g, respectively; for the turbine building, 0.5g and 0.7g. For valves in the containment-spray system, the value of a corresponding to a failure frequency of unity was taken to be 4.2g; for electrical switchgear, a was 0.67g. Similarly, the values of a and b for safety-injection piping were assumed to be 2.0g and 4.3g, respectively.

Consequence calculations were performed for both seismic and internal events, using the release-category frequencies reported in the Reactor Safety Study (USNRC, 1975). In both calculations, consequence models specific to the Diablo Canyon site were used. It was concluded that the seismic contribution to the overall radiological risk from this plant is low. The turbine building, which is not classified as a Seismic Category I structure and houses emergency diesel generators, switchgear, interface heat exchangers, and the fire-protection system, was found to be the source of most of the risk due to earthquakes.

11.2.2.2 Oyster Creek Seismic Risk Analysis

A seismic risk study was conducted as part of an overall safety study for the Oyster Creek plant (Garrick and Kaplan, 1980; Kennedy et al., 1980). Much of the development work for the seismic risk analysis that is discussed in this procedures guide and has been recently applied to several PRA studies was done in the Oyster Creek seismic risk study. A distinguishing feature of this study was the development and use of uncertainty estimates for both the ground-motion occurrence frequencies and the conditional frequencies of failure for structures and components. Since the results of this study have not yet been published, the discussion has to be limited to the overall approach and is covered in Section 11.2.11.

11.2.2.3 Big Rock Point PRA Study

The Big Rock Point PRA study evaluated the contribution of seismic events to the frequency of core melt (Consumers Power Company, 1981). The study consisted of a seismic hazard analysis, a component-fragility evaluation, and an assessment of different seismically induced accident sequences as to their contribution to the core-melt frequency. The seismic hazard analysis was performed with a model of tectonic zones in the Northern and Central United States (Algermissen and Perkins, 1976) and published attenuation and intensity-acceleration relationships. Component fragilities were evaluated by assigning the components to one of three categories: (1) the acceleration levels are sufficiently low that failure is not a problem; (2) the fragility of similar components is known or can be inferred; or (3) the seismic response of the component can be estimated for any earthquake ground acceleration, and the response can be compared against the capacity of the component.

From the evaluation of a limited number of components, the study concluded that the electrical components in the power room, control room, and reactor building are most vulnerable to seismically induced failures at peak ground accelerations of less than 0.20g. At larger earthquakes (i.e., accelerations greater than 0.20g), a collapse of the emergency condenser and core-spray failures induced by circuit-breaker trips or by a collapse of the turbine building are likely. The frequency of a seismically induced core melt was estimated to be 1.2×10^{-7} per year.

The study was somewhat limited in scope. The hazard analysis did not take into account all the uncertainties (e.g., uncertainty in attenuation, source modeling, and upper-bound magnitude), and in the calculation of component fragilities a composite variability (i.e., combination of inherent randomness and uncertainty) was used.

11.2.3 SEISMIC HAZARD ANALYSIS

Seismic hazard is usually expressed by the frequency distribution of the peak value of the ground-motion parameter during a specified interval of time. The major elements of this analysis (see Figure 11-3) are as follows:

1. Identification of the sources of earthquakes, such as faults (F1, F2) and seismotectonic provinces (A1, A2, A3) (sources).

2. Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities (recurrence).

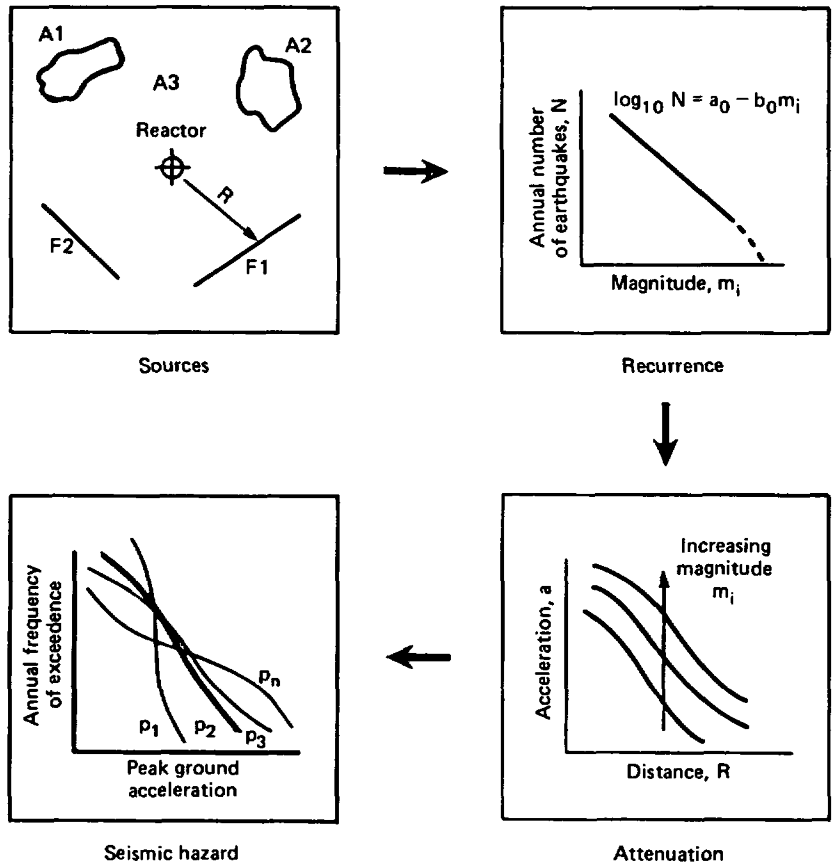

Figure 11-3. Model of seismic hazard analysis.

3. Development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site (attenuation).

4. Integration of all the above information to generate the frequencies with which different values of the selected ground-motion parameter would be exceeded (seismic hazard).

A comprehensive seismic hazard model was first proposed by Cornell (1968). Improvements to this model have been proposed by Cornell and Merz (1975), Shah et al. (1975), Algermissen and Perkins (1976), Der Kiureghian and Ang (1977), McGuire (1976), Mortgat et al. (1977), and Der Kiureghian (1981).* The next section describes the basic model with the modifications necessary for specific sites.

*Although most of these papers are called "seismic risk analysis procedures," they cover only the seismic hazard analysis as defined herein.

11.2.3.1 Seismic Hazard Model

An earthquake is a complex occurrence involving the geologic characteristics of the site, the buildup of crustal strains, slippage along fault planes, rupture surface, focus, and many other variables. The transmission of earthquake disturbance from the source to the site depends on the magnitude, distance, depth of focus, geologic characteristics of the region, type of fault movement, characteristics of earthquake waves (body and surface waves), etc. The ground motion at a site can be described by means of ground-motion parameters, such as the peak acceleration, velocity, displacement, the duration of motion, and a set of response-spectra amplitudes corresponding to the modal frequencies and dampings of the structure. Other possible descriptors are the Modified Mercalli (MM) intensity or energy-related ground-motion-intensity quantities, such as the root mean square of the ground acceleration or velocity.

Obviously, a complete description of the seismic hazard at a specific site should include all of the above variables for earthquake occurrence, wave transmission, and ground-motion input to the building. With its large number of variables, such a model would, in theory, lead to a more accurate estimate of seismic hazard, but it would be prohibitively complex. In practice, the modeling is kept tractable by retaining only a few dominant variables. The choice of a variable depends also on the type, the quality, and the amount of data. For example, the size of an earthquake is generally measured in terms of the Richter magnitude and epicentral intensity. Although other measures would perhaps be more appropriate (e.g., seismic moment and energy release), data on earthquakes have historically been gathered in terms of the Richter magnitude and the epicentral intensity on the Modified Mercalli scale.

The transmission of earthquake disturbance is generally represented in the hazard model by means of an attenuation relationship between a few significant variables (e.g., magnitude, epicentral intensity, distance, region, and soil type). The effect of other variables in the transmission process is accounted for by incorporating the observed scatter about the empirical attenuation relationship. Similarly, the earthquake ground motion may be characterized by a single parameter, such as the peak ground acceleration. The effect of other variables that are necessary for an adequate description of the ground motion may have to be included in the response analysis and fragility evaluation (e.g., using appropriate response spectra and recorded earthquake time histories).

The particular parameter or parameters that are chosen to present the results of a seismic hazard analysis depend on the plant-system and accident-sequence analysis. In the seismic risk studies done to date, the analysts have elected to characterize the seismic hazard in terms of the peak ground acceleration or some related parameter ("sustained" or "effective" peak ground acceleration).

The seismic hazard model is described in detail by Cornell (1968), Merz and Cornell (1973), and Cornell and Merz (1975). The model is used to calculate the annual mean number of events, $\nu_\ell(a)$, in which a ground-motion parameter A (e.g., peak ground acceleration) exceeds a value a at the site because of an earthquake on the ℓ^{th} seismic source as expressed below (Kulkarni et al., 1979):

$$\nu_\ell(a) = \sum_i \sum_j \nu_\ell f_\ell(m_i)\ f_\ell(r_j)\ f_{A|m_i,r_j}(a) \tag{11-1}$$

where

ν_ℓ = mean annual number of earthquakes on the source; called the activity rate of the source.

$f_\ell(m_i)$ = conditional frequency of the earthquake on the source having a magnitude* equal to m_i. The product $\nu_\ell f_\ell(m_i)$ is obtained from the well-known Gutenberg-Richter (1942) recurrence relationship $\log_{10} N = a_0 - b_0 m_i$, where N is the number of earthquakes per year exceeding magnitude† m_i; a_0 and b_0 are constants that depend on the seismicity of the region.

m_i = Richter or local magnitude of earthquake i.

$f_\ell(r_j)$ = frequency with which the source-to-site distance is r_j, given an earthquake on the ℓ^{th} source.

$f_{A|m_i,r_j}(a)$ = frequency with which the ground-motion parameter A exceeds the value a given an earthquake of magnitude m_i at a distance r_j.

The term $f_\ell(r_j)$ defines the location of the site with respect to the seismic source. Actually, the seismic source is divided into a number of discrete point sources, and the distances r_j are measured from the point sources to the site. The term $f_{A|m_i,r_j}(a)$ is a function of ground-motion attenuation from the source to the site.

By summing the contributions from all seismic sources around the site, the total annual mean number of events, $\nu(a)$, in which A exceeds a at the site can be obtained:

$$\nu(a) = \sum_\ell \nu_\ell(a) \tag{11-2}$$

The annual frequency of earthquakes in which the ground-motion parameter A is smaller than a is obtained by assuming that strong motions are Poisson events:

$$H(a) = e^{-\nu(a)} \tag{11-3}$$

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*The model is not limited to situations where magnitude data are available (see Section 11.2.3.2).

†In the original paper, N was the number of earthquakes with magnitude equal to m_i.

It is customary to plot the annual frequency of exceedence, $1 - H(a)$, as a function of the ground-motion parameter.

The annual frequency $h(a)$ of earthquakes in which the value of the ground-motion parameter A is between a and $(a + \Delta a)$ is given by

$$h(a) = H(a + \Delta a) - H(a) \tag{11-4}$$

This hazard estimate $h(a)$ depends on uncertain professional estimates of parameters, such as attenuation laws, upper-bound magnitudes, and the geometry of the source. Such uncertainties are included in the hazard analysis by assigning probabilities to alternative hypotheses about these uncertain parameters. A probability distribution for the frequency of occurrence is thereby developed.

The annual frequencies for exceeding specified values of the ground-motion parameter are displayed (see Figure 11-4) as a family of curves at different nonexceedence-probability levels.

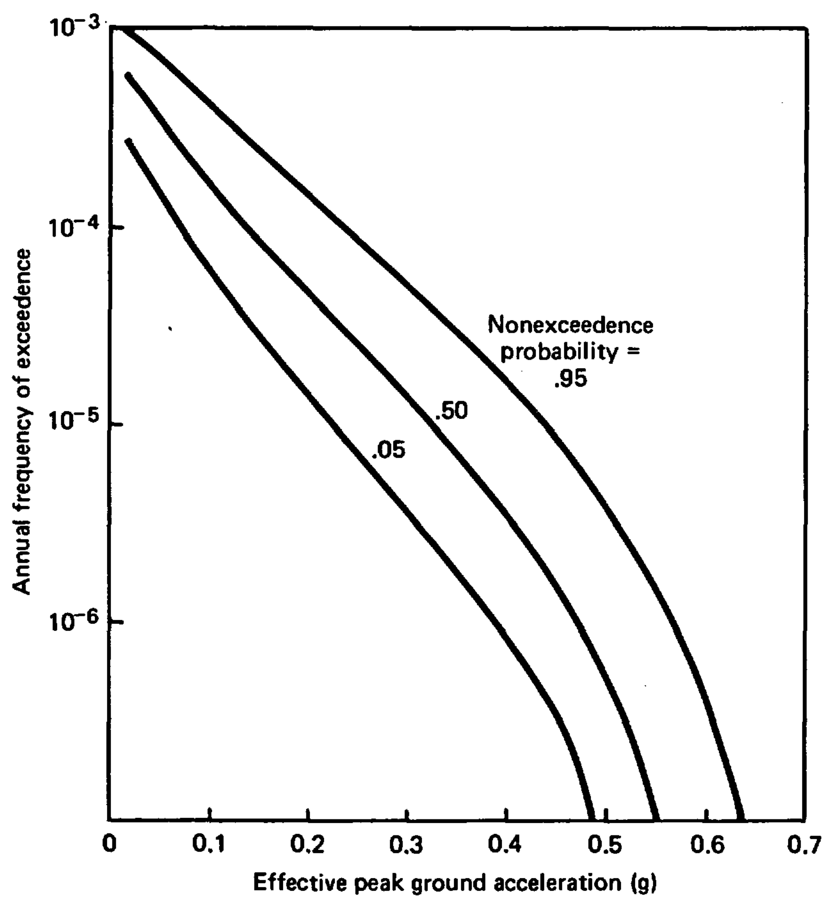


Figure 11-4. Seismic hazard curves for a hypothetical site.

11.2.3.2 Parameters of the Hazard Model

The parameters that characterize a seismic hazard model for a site are the seismic sources, the activity rate ν_ℓ, the relative frequencies of occurrence of different sizes of earthquakes on a source, the attenuation of ground motion from a source to the site, and the upper-bound magnitude or epicentral intensity of a source. In performing the seismic hazard analysis for a site, uncertainties in each of the above model parameters should be consistently treated, as explained in the sections that follow.

Seismic Sources

The seismic hazard model uses three types of seismic source: line source, area source, and point source. In fact, the numerical computation of the seismic hazard at the site is carried out by dividing the line and area sources into a number of discrete point sources. The line source is used to model faults or fault provinces. In a geographical region where recorded earthquakes cannot be related to any well-defined fault system, the concept of seismotectonic provinces is invoked, and the region is represented by a set of area sources, or seismogenic zones. A source, whether a line source or an area source, is distinguished by a uniform seismic activity; that is, the mean rate of earthquake occurrence per unit length or unit area is constant over the entire source.

The seismic sources around a site are identified by studying the epicentral locations of past earthquakes together with the geologic and geomorphologic features of the site region. The inclusion of a source in the hazard model depends on its contribution to the seismic hazard, which, in turn, depends on the activity rate, the upper-bound magnitude or epicentral Modified Mercalli (MM) intensity, and the distance to the site.

In defining the geometric configuration of a fault or a seismotectonic province, there are generally differences in interpretation among seismologists. Recent studies (TERA, 1979; McGuire, 1981) have focused on quantifying such differences in opinion through expressed degrees of belief in alternative geometric configurations for a seismic source.

Activity Rates

The mean annual rate of earthquakes over a source is known as the activity rate (ν_ℓ) of the source. This rate is estimated from the historical seismicity in that source. Historical earthquake data are generally available in magnitude or MM intensity values. It may be noted that there are many definitions of magnitude; examples are the local magnitude M (Gutenberg and Richter, 1954) and the body-wave magnitude m_b (Gutenberg, 1956). There are a number of empirical relationships between these two magnitude scales; for example, Brazee (1976) has proposed the following:

$$M = 1.34m_b - 1.71 \qquad (11\text{-}5)$$

In this chapter the term "magnitude" is used for measurement on both scales. Where a specific scale is intended, the corresponding definition and symbol are used.

Earthquakes of small magnitude (i.e., $m < 3.0$) or epicentral intensity (i.e., MM intensity < V) are not considered in estimating the activity rate because they rarely cause structural damage. For regions of high seismicity, the calculated activity rate for the fault or the seismogenic region can be considered to be stable; even for regions of low seismicity, such as the Midwestern and Eastern United States, it has been shown (McGuire, 1977a; McGuire and Barnhard, 1981) that the historical rates of seismic activity can be considered to be stable for the purposes of seismic hazard analysis. Any significant change in the seismic activity of a seismogenic region requires several centuries; therefore, the historical seismic activity is sufficiently representative for making hazard estimates for the operating life of a nuclear plant.

Relative Frequencies of Different Sizes of Earthquakes

The activity rate for a seismic source is the mean number of earthquake events regardless of their magnitudes (or intensities). The distribution of earthquakes according to their magnitudes is given by the recurrence relationship (Gutenberg and Richter, 1942)

$$\log_{10} N = a_0 - b_0 m_i \qquad (11\text{-}6)$$

where N is the number of earthquakes with magnitude equal to or greater than m_i for a given source and over a given interval of time. The conditional frequency of earthquakes of magnitude m_i on the source ℓ is expressed as

$$f_\ell(m_i) = \frac{\beta_0 \exp[-\beta_0(m_i - m_0)]}{\{1 - \exp[-\beta_0(m_m - m_0)]\}} \Delta m \quad (m_0 \leq m_i \leq m_m) \qquad (11\text{-}7)$$

where $\beta_0 = b_0 \ln 10$, m_0 is the magnitude below which earthquakes rarely cause structural damage (e.g., $m_0 = 3$), m_m is the upper-bound magnitude for the source, and Δm is the interval between magnitudes on the magnitude scale. The value of b_0 for a source is derived by plotting the logarithm of the number of recorded earthquakes that exceeded a particular magnitude against the magnitude and by fitting a linear relationship as in Equation 11-6. For regions of low seismicity, historical data may not be sufficient to develop a recurrence relationship like Equation 11-6 for each source zone. A single b_0 value may be appropriate for all source zones in the region. A typical average value of b_0 for the Eastern United States is 0.9 (McGuire, 1981).

Equivalent forms of Equations 11-6 and 11-7 can be developed when the earthquake data are on the MM intensity scale, which has been traditionally used to record most historical earthquakes in the Midwestern and Eastern United States. The recorded epicentral intensity I_0 is then converted to body-wave magnitude m_b. Cornell and Merz (1975) and McGuire (1977a,b) have shown how the seismic hazard analysis is performed with MM intensity data.

Upper-Bound Magnitude or Epicentral Intensity

The recurrence relationship given by Equation 11-6 predicts nonzero frequency of exceedence whatever the magnitude of the earthquake, but most seismologists believe that there is a physical limit on the size of earthquakes that can be generated by a seismic source. However, seismologists do not generally concur on a single value for this upper bound for any given source. For a well-defined fault, the upper-bound magnitude of the earthquake that the fault is capable of generating can be estimated from the rupture length (Tocher, 1958; Bonilla, 1970; Housner, 1970; Wallace, 1970; Mark, 1977).

It should be noted that the data used to derive these empirical relationships exhibit considerable scatter and that the relationship is markedly different for different regions of the world and for different types of fault movement (i.e., normal slip, reverse slip, strike slip, etc.). The hazard analyst should take into account this scatter about the mean relationships.

Wallace (1970) has developed a procedure for estimating earthquake-recurrence intervals (i.e., equivalently, frequencies of earthquakes of various magnitudes on a fault) from geologic evidence of long-term deformation rates for active faults. In a region where the recorded earthquakes cannot be correlated with any known faults, estimating the upper-bound magnitude or the epicentral intensity of a seismic source is best done by soliciting the opinion of experts (TERA, 1979). McGuire (1977a) has suggested that a probability distribution be assigned to different upper-bound magnitudes or epicentral intensities assumed for a source. Again, this probability distribution can be derived by analyzing the opinion of experts.

Attenuation

The decrease in the intensity of ground shaking with distance from the epicentral region is called "attenuation." Many empirical formulas have been proposed (see, for example, Donovan, 1973; Nuttli, 1973; Gupta, 1976; Murphy and O'Brien, 1977; McGuire, 1978; Campbell, 1981; Joyner and Boore, 1981). It has been observed that the attenuation of ground motion varies in different parts of the world. In the Western United States, earthquake motion attenuates more rapidly than it does in the Eastern or Midwestern United States. For the Western United States, where strong-motion instrumental data are available for a number of earthquakes, a typical attenuation formula has the form

$$a = b_1 \exp(b_2 m) \; (R + 25)^{-b_3} \qquad (11\text{-}8)$$

where a is the peak ground acceleration at the site (cm/sec^2), m is the magnitude of the earthquake (Richter or local magnitude), R is the distance to the energy center or the causative fault (km), and b_1, b_2, and b_3 are coefficients that are evaluated by using recorded strong-motion data. For example, Donovan and Borstein (1977) have reported the following values:

$$\begin{aligned} b_1 &= 2{,}154{,}000(R)^{-2.10} \\ b_2 &= 0.046 + 0.445 \log_{10} R \qquad (11\text{-}9) \\ b_3 &= 2.515 - 0.486 \log_{10} R \end{aligned}$$

An attenuation relationship like Equation 11-8 is generally a best fit to the data, which exhibit considerable scatter. This dispersion about the attenuation equation should be properly included in the hazard analysis. For example, Donovan and Borstein (1977) have reported the logarithmic standard deviation in the peak ground acceleration predicted by Equations 11-8 and 11-9 as ranging from 0.3 to 0.5.

Other empirical attenuation formulas developed for the Western United States include those by Schnabel and Seed (1972), McGuire (1974), Trifunac and Brady (1975), Blume (1977), Espinosa (1980), Campbell (1981), and Joyner and Boore (1981). The choice of any formula depends on the site geology, distance to active faults, and the availability of strong-motion data. Whichever formula is used, the analyst should take into account the dispersion in the data about the formula.

For the Eastern and Midwestern United States, where most recorded earthquake data are in MM intensity units, two approaches are available for specifying the attenuation of ground motion. In the first approach, the analyst begins by selecting an intensity-attenuation relationship appropriate for the region. An example of such an intensity attenuation is given by Gupta (1976) for the Central United States:

$$I_s = I_0 + 2.35 - 0.00316R - 1.79 \log_{10} R \quad (R \geq 20 \text{ km}) \qquad (11\text{-}10)$$

where I_s is the site intensity in MM units and I_0 is the epicentral intensity in MM units. The site intensity I_s is converted to the instrumental peak ground acceleration a_{pi} by using a relationship like that of Murphy and O'Brien (1977):

$$\log_{10} a_{pi} = 0.25I_s + 0.25 \qquad (11\text{-}11)$$

Because of the paucity of strong-motion data for the Eastern and Midwestern United States, it may be necessary to use the intensity-acceleration relationship developed from data for the Western United States, Japan, and Europe, such as Equation 11-11.

Equations relating the site intensity to the epicentral intensity and those relating the peak instrumental ground acceleration to the site intensity are best fits to the earthquake data, which normally exhibit wide scatter. For example, the standard deviation of site intensity about the predicted value of Equation 11-10 is reported as 0.5 MM intensity unit. Murphy and O'Brien (1977) have reported that the logarithmic standard deviation of the estimate associated with Equation 11-11 is 0.36.

In the second approach, the MM epicentral intensity I_0 is converted into the body-wave magnitude m_b by using an appropriate empirical relationship (Aggarwal and Sykes, 1978; Nuttli, 1979). For example, Nuttli (1979) gives

$$m_b = 0.5I_0 + 1.75 \qquad (11\text{-}12)$$

As before, the dispersion about this type of relationship (±0.5 magnitude unit for Equation 11-12) is to be included in the analysis.

The sustained maximum ground acceleration a_s is obtained for an earthquake of body-wave magnitude m_b at a distance R from the site by using a suitable attenuation relationship. For example, from Nuttli's theory (1979) McGuire (1981) has derived the following attenuation equation for the Central United States:

$$a_s = 0.584 \exp[-0.427 \exp(-0.444m_b) + 1.098m_b] \quad (R < 10 \text{ km})$$

$$a_s = 3.98R^{-5/6} \exp[-0.0427R \exp(-0.444m_b) + 1.098m_b] \quad (R \geq 10 \text{ km}) \tag{11-13}$$

McGuire (1981) also suggests a value of 0.6 as the logarithmic standard deviation about the mean value of a_s obtained with Equation 11-13.

The specific attenuation approach and the formulas used in the hazard analysis depend on the site region and the availability of intensity and strong-motion data. Cornell et al. (1979) have discussed the variabilities introduced in the predicted response by different attenuation approaches. Their nominal results for the standard deviation include, however, both natural dispersion and the systematic bias introduced by substitutions like Equation 11-10 into Equation 11-11.

In some seismic risk studies, such as the Zion PRA (Commonwealth Edison Company, 1981), the analysts have required that the seismic hazard be expressed in terms of the effective peak ground acceleration for compatibility with the component fragilities, which are derived in terms of the effective peak ground acceleration. Described below are two candidate procedures for expressing the seismic hazard in terms of the effective peak ground acceleration.

In the first procedure, the instrumental peak ground acceleration is reduced by an appropriate factor to obtain the sustained maximum ground acceleration a_s. The quantity a_s is defined as the level of acceleration corresponding to the third highest peak in the acceleration time history (Nuttli, 1979). Kennedy (1981) has suggested that the effective peak ground acceleration a_D can be taken as 1.25 times the sustained maximum ground acceleration.

The second procedure--proposed by McCann and Shah (1979), Mortgat (1979), and Vanmarcke and Lai (1980)--uses the root-mean-square acceleration A_{rms} as the ground-motion parameter of interest. The effective peak ground acceleration is related to the rms acceleration by

$$a_D = K_p A_{rms} \tag{11-14}$$

where K_p is a function of the acceptable exceedence frequency p for each individual peak of the time history:

$$K_p = \frac{\ln (1/p)}{\sqrt{2}} \tag{11-15}$$

Mortgat (1979) has shown how A_{rms} is used as the ground-motion parameter in a seismic hazard analysis. Whether this procedure can be used depends on the availability of strong-motion data, which are needed to develop attenuation relationships like Equation 11-8 in terms of the rms acceleration.

11.2.3.3 Other Models for Hazard Analysis

The basic hazard model described in Section 11.2.3.1 has been developed over the last 15 years and has had the benefit of improvement through specific applications (Donovan, 1973; Shah et al., 1975). At present, the hazard models actually used in specific site applications differ only in the parameter values. Several investigators have shown how the seismic hazard parameters can be evaluated by using Bayesian techniques to augment sparse data (Benjamin, 1968; Cornell and Vanmarcke, 1969; Esteva, 1969; Mortgat, 1976; Campbell, 1977; Eguchi and Hasselman, 1979; TERA, 1980) and using expert opinion (TERA, 1979).

The basic model assumes that earthquake events follow the Poisson model. The assumption that earthquakes are independent in time, as implied in this model, has been questioned by some seismologists. A Markovian assumption of one-step memory in time may be more valid, but the Poisson assumption for large events does not introduce major errors (Gardner and Knopoff, 1974). Der Kiureghian and Ang (1977) have proposed a line-source model that considers earthquakes originating as slips along geologic faults and assumes that the shortest distance to the slipped area is the important parameter in estimating the ground-motion intensity at the site. The use of this model for sites in regions with seismic activity concentrated on geologic faults may lead to better estimates of the seismic hazard.

11.2.3.4 Sensitivity Studies

Cornell and Vanmarcke (1969), Cornell and Merz (1975), Donovan and Borstein (1977), and McGuire (1977a, 1981) have studied the sensitivity of seismic hazard estimates to variations in the model-parameter values. The results of seismic hazard analysis are found to be especially sensitive to the mean attenuation function and to the dispersion about this function. Hence, the analyst must ascertain that the attenuation relationship is appropriate to the site region and that the scatter in the data is consistently accounted for.

An accepted procedure for including the uncertainties of the parameters in the hazard analysis is to postulate a set of hypotheses. Each hypothesis will consist of, for example, a specified configuration of the seismic sources, a value of the Gutenberg-Richter slope parameter b_0, a value of the upper-bound magnitude or epicentral intensity for each seismic source, and a cutoff value for the effective peak ground acceleration (Kennedy, 1981). A probability value is assigned to each of these hypotheses, based on the analyst's degree of belief and expert opinion. A seismic hazard curve representing the annual frequency of exceeding a specified effective peak ground acceleration is generated for each hypothesis. Some studies (see example, Cornell and Merz, 1975; TERA, 1980) have used such approaches in calculating the mean frequencies of exceeding different acceleration levels. For PRA applications, it is more appropriate to present the seismic hazard at the site as a family of hazard curves with different nonexceedence-probability levels (Figure 11-4).

11.2.3.5 Computer Codes

The following computer codes can be used in analyzing the seismic hazard at the site of a nuclear power plant:

1. SRA (Seismic Risk Analysis), developed by C. A. Cornell at the Massachusetts Institute of Technology, 1975.

2. EQRISK, a Fortran code developed by McGuire (1976). It is available from the National Information Service for Earthquake Engineering, University of California, Berkeley.

3. FRISK, a code for seismic risk analysis using faults as earthquake sources, developed by McGuire (1978).

4. Seismic Risk Analysis Program by C. P. Mortgat, the John A. Blume Earthquake Engineering Center, Stanford University, Stanford, California, 1978.

5. HAZARD, developed at the Lawrence Livermore National Laboratory, Livermore, California, 1980.

11.2.3.6 Case Studies

Cornell and Merz (1975) have described the analysis of the seismic hazard at a site in the Eastern United States. They discuss the process of selecting the parameter values and the sensitivity of hazard estimates to variations in these values. Applications to sites in the Western United States that are exposed to line sources have been described by Shah et al. (1975) and Donovan and Borstein (1977). For recent applications in seismic risk studies for nuclear power plants, the reader is referred to the Zion PRA (Commonwealth Edison Company, 1981) and to reports by TERA (1980) and Chung and Bernreuter (1981).

11.2.4 ANALYSIS OF PLANT-SYSTEM AND STRUCTURE RESPONSES

In order to calculate failure frequencies for structures, equipment, and piping, it is necessary to obtain the seismic responses of these components to various levels of the ground-motion parameter (e.g., peak ground acceleration). The breadth and depth of the response analysis depend on the information existing on analyses performed during the design stage and on the method used to develop component fragilities. For older nuclear plants (those built in the 1960s), seismic design procedures and criteria would have been much different from the current ones, and not all of the seismic design information (e.g., structural and piping analysis models, stress reports, and equipment-qualification reports) may be available. For such plants, it may be necessary to develop structural and piping analysis models and to calculate the responses for critical components. Some amount of iteration and interaction between the structures analyst and the systems analyst would reduce the amount of response analysis by concentrating on the critical structures and components. For a "newer" plant, the analyst can rely on the design-analysis information.

If a detailed response analysis is needed, the following procedure is used. Design drawings and as-built conditions are reviewed to develop structural analysis models for the critical structures. If the analyst thinks that the effects of soil-structure interactions are important, such effects can be incorporated by using a direct method that models the soil and the structure together or by using a substructure approach (Johnson, 1980). Since correlations between the components are needed to estimate the joint failure frequencies for a set of components in an accident sequence, structure and piping-system analyses are performed by means of time-history methods. The variability in the input ground motion is incorporated by simulating a set of time histories consistent with the hazard curve. For example, in the Seismic Safety Margins Research Program (SSMRP--Smith et al., 1981), a hazard model was developed in order to select a set of time histories each of which simulated a particular peak spectral acceleration and spectral shape. Subsystem responses (i.e., piping responses) are determined by using a multisupport time-history analysis. The subsystems may consist of valves, nozzles, and pumps as well as piping nodes.

Although some component failures may involve inelastic responses, most current analyses are limited to linear dynamic analyses of structures and subsystems. Nonlinear response effects are accounted for by estimating the inelastic-energy-absorption capacity for the component under study; the ductility-factor approach of Newmark (1977) is used for this estimate.

The output of the response analysis is the frequency density function of the peak response (e.g., moment, stress, and deformation) of each critical component and the covariances between component responses. Variabilities in the input parameters (e.g., soil shear modulus and damping, and structure and subsystem response frequencies and damping) are incorporated by using an appropriate sampling technique, such as the Latin hypercube (Iman et al., 1980). By separating the variability of each parameter into randomness and uncertainty, and by assigning probabilities to express uncertainties, a probability distribution on the cumulative distribution of component response is derived.

Alternatively, the analyst may decide to estimate the actual component response for a given level of seismic input from the available design-analysis information. A response factor of safety is derived from a linear dynamic analysis of the structure or equipment. In most cases, the response factor of safety can be estimated from the results of response analyses performed for the design-earthquake levels (e.g., operating-basis and safe-shutdown earthquakes) and ground-response spectra. This factor of safety depends on the safety factors involved in the selection of ground-response spectra, the procedure used to include the effects of soil-structure interactions, the selected damping levels, the modeling of structures and piping, and the method of analysis. The safety factors are treated as random variables, and their statistical parameters, such as the median and the logarithmic standard deviation, are estimated by using available data and engineering judgment.

While this approach circumvents the need for a detailed response analysis, it does consider important variables that might affect the responses of structures and equipment. It is expected that the overall variability in response predicted by this approach will be higher than that obtained by a detailed modeling and analysis of structures. Furthermore, any correlation between component responses can be treated only approximately because in the safety-factor evaluation approach equipment, structural elements, piping, cable trays, etc., are examined separately and not as an assemblage.

The responses calculated as described above are related to the responses of structural elements, piping, and any on-line equipment (e.g., valves, nozzles, and pumps). Some equipment that is mounted on the floor or attached to walls may not be included in the dynamic analysis models of main structures and subsystems. The structural analysis will yield the floor spectra (more specifically, a frequency distribution for the floor spectra) for the particular equipment. The actual response of the equipment depends on its dynamic characteristics and how it is qualified.

The frequency density function of the equipment response is derived by modifying the floor-response frequency density function by a multiplicative factor called the equipment-response factor F_{RE}. This factor is a random variable that accounts for variabilities due to (1) the equipment-qualification method, (2) modeling error (i.e., frequency and mode shape), (3) damping, (4) modal response combination, and (5) earthquake-component combination. As before, the equipment-response factor is described by a set of frequency density functions, each with an assigned probability value.

11.2.4.1 Computer Code

The Seismic Safety Margins Research Program developed a computer code called SMACS (Seismic Methodology Analysis Chain with Statistics) for calculating the seismic responses of structures, systems, and components. This code links the seismic input in the form of ensembles of acceleration time histories with the calculations of soil-structure interactions, the responses of major structures, and the responses of subsystems. Since SMACS uses a multisupport approach to perform the time-history response calculations for piping subsystems, the correlations between component responses can be handled explicitly.

11.2.5 FRAGILITY EVALUATION

The fragility of a component is defined as the conditional frequency of its failure given a value of the response parameter, such as stress, moment, and spectral acceleration.

11.2.5.1 Failure Modes

The first step in generating fragility curves like those in Figure 11-5 is to develop a clear definition of what constitutes failure for each component. This definition of failure must be acceptable to both the structural analyst, who generates the fragility curves, and the systems analyst, who must judge the consequences of a component's failure in estimating plant risk. It may be necessary to consider several modes of failure (each with a different consequence), and fragility curves are required for each mode.

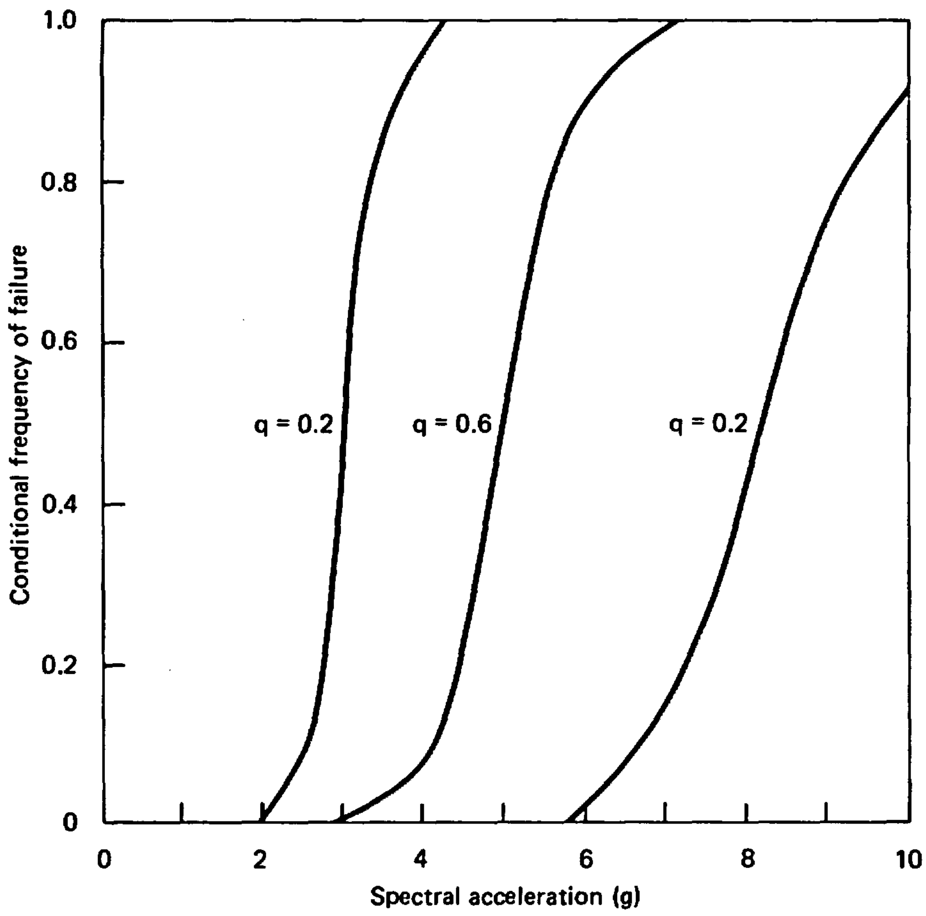


Figure 11-5. Fragility curves for a component.

For example, a motor-operated valve can fail in any of the following ways (Kennedy et al., 1980):

1. Failure of power or controls to the valve (generally related to the seismic capacity of the cable trays, control room, and emergency power). These failure modes are most easily handled as failures of separate systems linked in series to the equipment since they are not related to the specific piece of equipment (i.e., a motor-operated valve) and are common to all active equipment.

2. Failure of the motor.

3. Valve binding due to distortion.

4. Rupture of the pressure boundary.

By reviewing the equipment design, it may be possible to identify the failure mode that is most likely to be caused by the seismic event and to consider only that mode. Otherwise, in developing fragility curves it is necessary to use the premise that the component can fail in any one of all potential failure modes.

The identification of credible modes of failure is based largely on the analyst's experience and judgment. A review of plant design criteria, calculated stress levels in relation to the allowable limits, the results of qualification tests, and failures reported in licensee event reports, including fragility tests, is useful in this task. Piping, electrical, mechanical, or electromechanical equipment vital to the safety of a plant is considered to fail when it cannot perform its designated function. In some PRA studies, relay chatter and trip were considered to temporarily interrupt the component function or considered to be corrected manually. It was therefore assumed that the electrical components would not fail in this mode. However, in judging relay failures as recoverable, the analyst should consider the possibility of spurious alarms and incorrect actions by the operator. For piping, a failure of the support system or a plastic collapse of the pressure boundary is considered to be the dominant failure mode.

Structures can be considered to fail functionally when the inelastic deformations under seismic loads are estimated to be sufficient to potentially interfere with the operability of safety-related equipment attached to the structure or fractured sufficiently for equipment attachments to fail. These failure modes represent a conservative lower bound on the seismic capacity because a nuclear plant structure has a considerably greater margin of safety against collapse. However, a structural collapse should generally be assumed to result in the failure of all safety-related equipment or systems housed inside the portion of the structure that is judged to have failed; that is, the structural failure results in a common-cause failure of multiple safety systems if they are housed in the same structure. The event and fault trees should appropriately reflect this condition.

Consideration should also be given to the potential for soil failure in various modes: liquefaction, toe-bearing pressure failure, slope failures, and base-slab uplift. For buried structures (i.e., piping and tanks), failure due to lateral soil pressures may be important. Both structures and equipment may be damaged through an earthquake-induced impact by another structure or equipment (e.g., a crane). A seismically induced dam failure, if any dams are present nearby, should also be investigated, as it may result in flooding or a loss of cooling source.

11.2.5.2 Calculation of Component Fragilities

Component fragility, which is defined as the conditional frequency of failure for a given value of the response parameter, is calculated by developing the frequency distribution of the seismic capacity of a component and finding the frequency for this capacity being less than the response-parameter value.

Seismic Capacity

The seismic capacity of a structure, piping system, or a piece of equipment is calculated by considering both the strength (i.e., ultimate strength or strength at loss of function) and the capacity for inelastic energy absorption; the latter term refers to the fact that an earthquake is a limited energy source, and many structures and equipment are capable of absorbing, without loss of function, substantial amounts of energy beyond yield.

In estimating the seismic capacity of a component, it is convenient to work in terms of an intermediate random variable called the capacity factor of safety, F_C. This factor is defined as the ratio of the capacity of the component to the magnitude of the fragility (local response) parameter specified for the reference earthquake (e.g., safe-shutdown earthquake), A_{SSE}. The quantity F_C is expressed as

$$F_C = F_S F_\mu \tag{11-16}$$

where F_S is the strength factor--that is, the ratio of ultimate strength (or strength at loss of function) to the stress calculated for A_{SSE}--and F_μ is the inelastic-energy-absorption factor, whose evaluation is discussed later in this section. For active components, the operability limits are likely to govern, and hence the median value of F_μ may be smaller than it is for structures. In calculating the strength factor F_S, the nonseismic portion of the total load (stress) or response acting on the component is subtracted from the strength, as shown below.

$$F_S = \frac{S - P_N}{P_T - P_N} \tag{11-17}$$

where S is the strength of the component, P_N is the normal operating load (stress), and P_T is the total load on the component--that is, the sum of the seismic load (SSE) and the normal operating load. For higher levels of earthquake, other transients (e.g., the discharge of safety relief valves and turbine trip) may have a high likelihood of occurring simultaneously with the earthquake and then the definition of P_N will be extended to include the loads from these transients. In combining the dynamic responses from earthquakes and transients to calculate P_N and P_T, a realistic procedure like the method of the square root of the sum of squares (SRSS) is used.

Sometimes, the strength S of a component is expressed as a function of a number of variables. For example, the shear strength of a concrete shear wall is a function of the compressive strength of the concrete, the yield strength of steel, the steel reinforcement ratio, and the like. The mean and standard deviation of the strength can be calculated by using first-order approximations (Benjamin and Cornell, 1970).

A complete description of the seismic capacity should include the variabilities due to both inherent randomness and uncertainty in the parameters of the model for capacity. Therefore, the appropriate seismic capacity for a specific failure mode is described by a set of frequency density functions (representing inherent randomness), each with an assigned probability q_i (representing the uncertainty in the parameter values). A model that considers both types of variability is explained below (Kennedy et al., 1980). The seismic capacity C is expressed as

$$C = \check{C}\, \varepsilon_{C,R}\varepsilon_{C,U} \tag{11-18}$$

where C is the median capacity, $\varepsilon_{C,R}$ is a random variable reflecting the inherent randomness in the capacity, and $\varepsilon_{C,U}$ is a random variable reflecting the uncertainty in the calculation of C. Both $\varepsilon_{C,R}$ and $\varepsilon_{C,U}$ are assumed to be lognormally distributed with unit median and logarithmic standard deviations $\beta_{C,R}$ and $\beta_{C,U}$, respectively. Recalling that the seismic capacity is expressed as the capacity factor of safety F_C times the reference value of the fragility parameter A_{SSE}, we use Equation 11-16 and the properties of the lognormal distribution to obtain

$$\check{F}_C = \check{F}_S\check{F}_\mu \tag{11-19}$$

$$\beta_{C,R} = \left(\beta^2_{S,R} + \beta^2_{\mu,R}\right)^{1/2} \tag{11-20}$$

$$\beta_{C,U} = \left(\beta^2_{S,U} + \beta^2_{\mu,U}\right)^{1/2} \tag{11-21}$$

where $\check{F}_C$, $\check{F}_S$, and $\check{F}_\mu$ are the median values of F_C, F_S and F_μ, respectively, $\beta_{S,R}$ and $\beta_{\mu,R}$ are the logarithmic standard deviations reflecting the inherent randomness in F_S and F_μ, and $\beta_{S,U}$ and $\beta_{\mu,U}$ are the logarithmic standard deviations reflecting the uncertainties in the median value of F_S and F_μ.

The inelastic-energy-absorption factor F_μ is a function of the allowable ductility ratio μ. Newmark and Hall (1978) have suggested that, for frequencies within the amplified acceleration range of the ground-response spectrum, the factor F_μ on capacity can be estimated by

$$F_\mu \approx (2\mu - 1)^{1/2}\varepsilon \tag{11-22}$$

where ε is a random variable to account for the uncertainty associated with the use of Equation 11-22 to define F_μ. The quantity ε is assumed to be lognormally distributed with unit median and logarithmic standard deviation β_ε, which is estimated to be 0.20 (Kennedy et al., 1980). The median ductility ratio $\check{\mu}$ and the logarithmic standard deviation β_d for the component are estimated from a review of the relevant literature and experimental data. Equation 11-22 and the values of $\check{\mu}$, β_d, and β_ε are then used to calculate the median and logarithmic standard deviations of F_μ.

The frequency distribution for seismic capacity is now developed from the values of $\check{C}$, $\beta_{C,R}$, and $\beta_{C,U}$. Figure 11-5 shows an example of fragility curves for a component. Plotted are the conditional frequencies of failure versus the spectral acceleration. The conditional frequency of failure at any given spectral acceleration is the frequency that the seismic capacity is less than or equal to the spectral acceleration. This is calculated from the parameters $\check{C}$, $\beta_{C,R}$, and $\beta_{C,U}$ along with the lognormal-distribution assumption. A set of fragility curves is developed. To each curve, a probability value q_i is assigned to reflect the uncertainty in the seismic capacity parameters.

The seismic capacities of some components may be correlated because the components are supplied by the same manufacturer or mounted in the same way. In such situations, a correlation-coefficient matrix of seismic capacities may be developed in order to calculate the joint failure frequencies of components in an accident sequence (Collins and Hudson, 1981). However, this is not done in routine PRA studies for lack of data on such correlations.

11.2.5.3 An Alternative Formulation of Component Fragility

In this formulation, the fragility of a component is expressed as the conditional frequency of failure for a given peak ground acceleration. Data on seismically induced fragilities are generally not available for equipment and structures. Fragility curves must therefore be developed primarily from analysis supplemented with engineering judgment and limited test data. In view of this, maximum use is made of the response-analysis results obtained at the plant design stage.

The component fragility for a particular failure mode is expressed in terms of the ground-acceleration capacity A. The fragility is therefore the frequency at which the random variable A is less than or equal to a specified value, a. The ground-acceleration capacity is, in turn, modeled as

$$A = \check{A}\ \varepsilon_{A,R}\varepsilon_{A,U} \tag{11-23}$$

where $\check{A}$ is the median ground-acceleration capacity, $\varepsilon_{A,R}$ is a random variable (with unit median) representing the inherent randomness about A, and $\varepsilon_{A,U}$ is a random variable (with unit median) representing the uncertainty in the median value.

It is assumed that both $\varepsilon_{A,R}$ and $\varepsilon_{A,U}$ are lognormally distributed with logarithmic standard deviations of $\beta_{A,R}$ and $\beta_{A,U}$, respectively. The advantages of this formulation are as follows:

1. The entire fragility curve and its uncertainty can be expressed by only three parameters: $\check{A}$, $\beta_{A,R}$, and $\beta_{A,U}$. With the limited data

available on component fragility, it is necessary to estimate only three parameters rather than the entire shape of the fragility curve and its uncertainty.

2. The product form in Equation 11-23 and the lognormal-distribution assumption make the fragility computations mathematically tractable.

The lognormal distribution can be justified as reasonable (Kennedy et al., 1980) because it can adequately represent the statistical variation of many material properties and seismic response variables, provided one is not primarily concerned with the extreme tails of the distribution. In addition, the central limit theorem states that a distribution of a random variable consisting of products and quotients of several variables tends to be lognormal even if the distributions of the individual variables are not lognormal. For estimating failure frequencies on the order of 1 percent or higher, this distribution is considered to be reasonably accurate. However, if the lognormal distribution is used for estimating the very low failure frequencies associated with the tails of the distribution, the approach is considered to be conservative: the low-frequency (probability) tails of the lognormal distribution generally extend farther from the median than the actual structural resistance or response data might extend since the data on material strength or response show cutoff limits beyond which there is essentially zero frequency of occurrence.

Using Equation 11-23 and the lognormal-distribution assumption, the fragility (i.e., the frequency of failure, f') at any nonexceedence probability level Q can be derived as

$$f' = \Phi\left[\frac{\ln(a/\check{A}) + \beta_{A,U}\,\Phi^{-1}(Q)}{\beta_{A,R}}\right] \tag{11-24}$$

where $Q = P(f < f'|a)$ is the probability that the conditional frequency f is less than f' for a peak ground acceleration a. The quantity $\Phi(\cdot)$ is the standard Gaussian cumulative distribution function, and $\Phi^{-1}(\cdot)$ is its inverse. For displaying the fragility curves, the nonexceedence-probability level Q is used. Subsequent computations are made easier by discretizing the probability distribution of frequency, Q, into values q_i associated with different values of the failure frequency f. A family of fragility curves, each with an associated probability q_i, is developed.

For example, let the fragility parameters of a component be $\check{A} = 0.73g$, $\beta_{A,R} = 0.30$, and $\beta_{A,U} = 0.28$; then, from Equation 11-24, the conditional failure frequency that is not exceeded with a 95-percent probability for a ground acceleration of 0.5g is found to be 0.60. At a 90-percent nonexceedence probability, the conditional failure frequency for a ground acceleration of 0.5g is approximately 0.52.

In estimating the fragility parameters, it is convenient, as before, to work in terms of an intermediate random variable known as the factor of safety F. This is defined as the ratio of the ground-acceleration capacity A to the safe-shutdown-earthquake (SSE) acceleration used in plant design.

The median factor of safety $\check{F}$ can be directly related to the median ground-acceleration capacity $\check{A}$ as

$$\check{F} = \frac{\check{A}}{SSE} \tag{11-25}$$

The logarithmic standard deviations $\beta_{F,R}$ and $\beta_{F,U}$ for F are identical with those for the ground-acceleration capacity.

For structures, the factor of safety is modeled as the product of three random variables:

$$F = F_S F_\mu F_{RS} \tag{11-26}$$

where F_S is the strength factor, F_μ is the inelastic-energy-absorption factor, and F_{RS} is the structure-response conservatism factor. The strength factor represents the ratio of the ultimate strength (or strength at loss of function) to the computed response level. The structure-response factor recognizes the variability in (1) ground motion and the associated ground-response spectra for a given peak acceleration, (2) soil-structure interactions, (3) energy dissipation (damping), (4) structural modeling, (5) the method of analysis, (6) the combination of modes or time-history analysis results, and (7) the combination of earthquake components.

For equipment and other components, the factor of safety is modeled as

$$F = F_S F_\mu F_{RE} F_{RS} \tag{11-27}$$

The factors F_S and F_μ together represent the capacity factor of safety for the equipment relative to the floor acceleration used for the equipment design. The factor F_{RE} represents the safety inherent in the computation of equipment response, and F_{RS} is the factor of safety in the structure-response analysis that resulted in the floor spectra for equipment design.

Median, $\check{F}_{(.)}$, and variability, $\beta_{(.)R}$ and $\beta_{(.)U}$, estimates are made for each of the parameters affecting the capacity and response factors of safety. Using the properties of the lognormal distribution, these median and variability estimates are then combined to obtain the overall median factor of safety $\check{F}$ and the variability, $\beta_{F,R}$ and $\beta_{F,U}$, estimates required to define the fragility curve for a structure or a component. It should be noted that $\beta_{(.)R}$ represents the sources of dispersion in the factor of safety that cannot be reduced by a more detailed evaluation or by gathering more data. These sources include but are not limited to (1) the variability in an earthquake time history and thus in structure response when the earthquake is defined only in terms of the peak ground acceleration and (2) the variability in material properties (structure, soil, and equipment), such as strength, inelastic energy absorption, and damping.

The dispersion represented by $\beta_{(.)U}$ arises from (1) the variability due to an insufficient understanding of structural material properties, (2) errors in the calculated response that result from using approximate modeling for the structure and inaccuracies in mass and stiffness representations, and (3) the use of engineering judgment in lieu of complete plant-specific data on the fragility levels of equipment and on responses.

Examples showing how fragility curves are developed for structures and equipment can be found in a paper by Kennedy et al. (1980) and in the Zion PRA study (Commonwealth Edison Company, 1981).

Although fragility curves are developed independently for different components, some dependence is likely to exist between the earthquake-induced failures of components, particularly for structural elements and equipment located inside the structures. This is so even though the failure events are conditional on the peak ground acceleration. If the components are on the same elevation of the structure, are made by the same manufacturer, and are oriented in the same direction, then perfect dependence between them may be assumed; if none of these conditions are met, then perfect independence may be assumed. However, if one or two of these three conditions exist, the analyst, lacking other means to establish the extent of dependence, may assume independence or dependence, whichever gives conservative results. In some instances, these assumptions may result in large dispersions in the plant-risk estimates, which would require further in-depth studies and modeling (Smith et al., 1981).

The information required for estimating component fragilities includes as-built layouts and dimensions of members; material-strength test data for concrete, reinforcing steel, and structural steel; plant design bases; design calculations; stress reports; and qualification procedures and test reports for equipment. A more detailed list of the information needed for developing component fragilities is given in Section 11.2.12. For mechanical and electrical equipment, fragility curves are based on design-analysis data, shock-test results (i.e., by the U.S. Army Corps of Engineers), and expert opinion (Vagliente, 1981).

11.2.5.4 Selection of Components for Response and Fragility Evaluation

The selection of components or systems for fragility evaluation is an iterative process and calls for a close interaction between the systems analyst and the structural analyst. The systems analyst provides a list of structures, systems, and components whose failure may lead to radiological consequences. He may be guided in this selection by the accident sequences identified for the internal events and by other published seismic risk studies (e.g., Diablo Canyon, Zion, SSMRP, and Big Rock Point). For a typical nuclear plant, this list may include from about 100 up to 300 components, depending on the detail employed in the plant-system and sequence analysis. In some studies (Smith et al., 1981; Commonwealth Edison Company, 1981) it has been necessary to group the equipment into generic categories. The structural analyst develops the response frequency distributions and fragility curves for significant failure modes for each of these structures, systems, and equipment. After reviewing plant design criteria, stress reports, and equipment-qualification reports and performing a walk-through inspection of the plant, he may add to, or delete from, the list of components.

In the process of developing fragility curves, the structural analyst may identify components that have low fragilities even at extremely high ground accelerations (e.g., six to eight times the SSE acceleration); for these components, further refinements in the form of detailed response analyses and data collection may not significantly influence the calculations of seismic risk. However, such refinements may be necessary for those components that are calculated to have significant frequencies of failure at ground accelerations of 1.5 to 3 times the SSE acceleration. The need for a detailed fragility evaluation finally rests on the significance of the components in an accident sequence and the contribution of that sequence to the plant seismic risk. By the procedure described here, the analysts in some PRA studies have reduced the number of major components in the plant logic for seismic events to as few as 10.

11.2.6 PLANT-SYSTEM AND ACCIDENT-SEQUENCE ANALYSIS

The frequencies of core melt and radionuclide releases to the environment are calculated by using the plant logic combined with component fragilities and seismic hazard estimates. Event and fault trees are constructed to identify the accident sequences that may lead to core melt and a release of radionuclides.

In the performance of plant-system and accident-sequence analysis, the major differences between seismic and internal events are in--

1. The identification of initiating events.

2. The increased likelihood of multiple failures of safety systems requiring a more detailed event-tree development.

3. More pronounced dependences between component failures as a result of correlation between component responses and between capacities.

11.2.6.1 Initiating Events

The first step in the plant-system and accident-sequence analysis is the identification of earthquake-induced initiating events. To this end, the initiating events postulated for the internal events (see Chapter 3) are reviewed to identify those that are relevant to the seismic risk study. For example, the following initiating events are used in the Seismic Safety Margins Research Program for a PWR plant (Smith et al., 1981):

1. Reactor-vessel rupture.

2. Large LOCA (rupture of a pipe larger than 6 inches in diameter or the equivalent).

3. Medium LOCA (rupture of a pipe 3 to 6 inches in diameter or the equivalent).

4. Small LOCA (rupture of a pipe 1.5 to 3.0 inches in diameter or the equivalent).

5. "Small-small" LOCA (rupture of a pipe 0.5 to 1.5 inches in diameter or the equivalent).

6. Transient with the power-conversion system (PCS) operable.

7. Transient with the PCS inoperable.

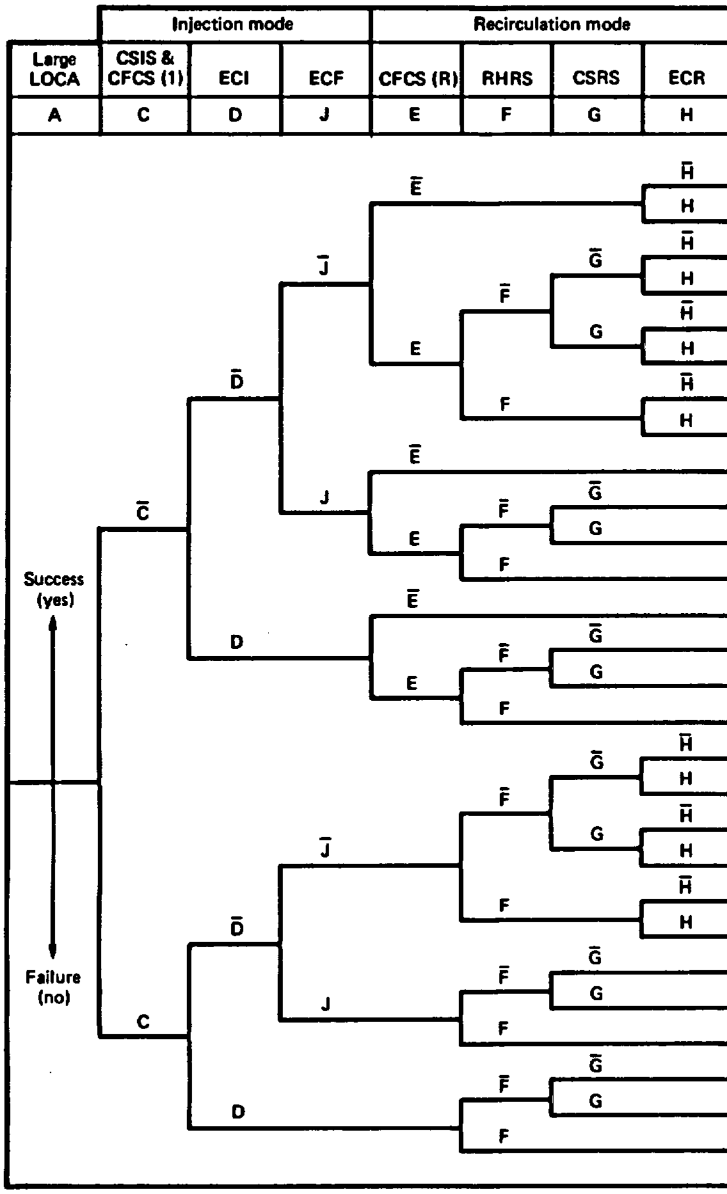
The conditional frequency of each initiating event is calculated for different levels of earthquakes. These levels are established on the basis of ranges in the peak ground acceleration (e.g., 0 to 0.15g, 0.15g to 0.30g, 0.30g to 0.45g). Given an earthquake in the acceleration range 0.15g to 0.30g, the analyst calculates a joint frequency distribution of the responses at different critical locations in the piping whose failure would lead to an initiating event (e.g., small LOCA). The convolution of the frequency distribution with the fragilities yields the conditional frequency of the initiating event.

In identifying the initiating events caused by earthquakes, the analyst may have to look beyond the single initiating events studied for internal events. For large earthquakes, multiple initiating events may occur at the same time, with markedly different effects on the engineered safety features (ESFs). For example, when a small LOCA that occurs without a complete blowdown is coupled with a loss of main feedwater, the effects on the capability of certain ESFs may be different from those of a loss of main feedwater or a small LOCA occurring as different events separated in time (Collins and Hudson, 1979). Another example would be the simultaneous break of a main-steam line and a LOCA. The inclusion of these and other initiating events in event trees depends on their conditional frequencies. Once the dominant initiating events have been identified, they can be arranged into a hierarchical order and grouped as described in Chapter 3.

11.2.6.2 Event Trees

The development of event trees for earthquake-induced initiating events follows essentially the methods described in Chapter 3. From these event trees, core-melt accident sequences are identified. Each of these core-melt sequences is followed by a containment sequence that establishes the release sequence. Figure 11-6 shows an event tree for a large LOCA in a PWR (Smith et al., 1981); it contains 23 core-melt sequences, and each sequence can lead to a release through the potential containment-failure modes designated α, β, γ, δ, and ε.

In developing the event trees, the analyst should be aware of the increased likelihood of multiple failures of safety systems under earthquake conditions. The systems that are essentially guaranteed to be available for mitigating accidents initiated by internal events may fail under earthquake conditions. For example, in a risk study for a PWR, the analyst may judge that, since three of five containment fan coolers will provide sufficient cooling for the containment in the event of core melt, the fan-cooler system is always available for mitigation. But a large earthquake may damage all five fan coolers, and this possibility has to be reflected in the seismic event trees. Also, if the failure of an ESF can lead to an initiating event, that ESF cannot appear on the corresponding event tree as available



Sequence	Seq.	Containment failure mode	Reactor Safety Study equivalent
$A\ \bar{C}\ \bar{D}\ \bar{J}\ \bar{E}\ \bar{H}^*$	1		
$A\ \bar{C}\ \bar{D}\ \bar{J}\ \bar{E}\ H$	2	$\alpha\beta\epsilon$	AH
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ \bar{G}\ \bar{H}^*$	3		
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ \bar{G}\ H$	4	$\alpha\beta\epsilon$	AH
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ G\ \bar{H}^*$	5		
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ G\ H$	6	$\alpha\beta\gamma\delta\epsilon$	AHF
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ F\ \bar{H}$	7	$\alpha\delta$	AG
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ F\ H$	8	$\alpha\beta\delta\epsilon$	AHG
$A\ \bar{C}\ \bar{D}\ J\ \bar{E}$	9	$\alpha\beta\epsilon$	AE
$A\ \bar{C}\ \bar{D}\ J\ E\ \bar{F}\ \bar{G}$	10	$\alpha\beta\epsilon$	AE
$A\ \bar{C}\ \bar{D}\ J\ E\ \bar{F}\ G$	11	$\alpha\beta\delta\epsilon$	AEF
$A\ \bar{C}\ \bar{D}\ J\ E\ F$	12	$\alpha\beta\delta\epsilon$	AEG
$A\ \bar{C}\ D\ \bar{E}$	13	$\alpha\beta\epsilon$	AD
$A\ \bar{C}\ D\ E\ \bar{F}\ \bar{G}$	14	$\alpha\beta\epsilon$	AD
$A\ \bar{C}\ D\ E\ \bar{F}\ G$	15	$\alpha\beta\epsilon$	AF
$A\ \bar{C}\ D\ E\ F$	16	$\alpha\beta\delta\epsilon$	ADG
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ \bar{G}\ \bar{H}^*$	17		
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ \bar{G}\ H$	18	$\alpha\beta\epsilon$	ACH
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ G\ \bar{H}^*$	19		
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ G\ H$	20	$\alpha\beta\epsilon$	ACH
$A\ C\ \bar{D}\ \bar{J}\ F\ \bar{H}$	21	$\alpha\delta$	ACG
$A\ C\ \bar{D}\ \bar{J}\ F\ H$	22	$\alpha\beta\delta\epsilon$	ACHG
$A\ C\ \bar{D}\ J\ \bar{F}\ \bar{G}$	23	$\alpha\beta\epsilon$	ACE
$A\ C\ \bar{D}\ J\ \bar{F}\ G$	24	$\alpha\beta\epsilon$	ACE
$A\ C\ \bar{D}\ J\ F$	25	$\alpha\beta\delta\epsilon$	ACEG
$A\ C\ D\ \bar{F}\ \bar{G}$	26	$\alpha\beta\epsilon$	ACD
$A\ C\ D\ \bar{F}\ G$	27	$\alpha\beta\epsilon$	ACD
$A\ C\ D\ F$	28	$\alpha\beta\epsilon\delta$	ACDG

Acronyms: CSIS, containment spray injection system; CFCS, containment fan cooler system (I = injection; R = recirculation); ECI, emergency coolant injection; ECF, emergency core-cooling function; RHRS, residual heat removal system; CSRS, containment spray recirculation system; ECR, emergency core-cooling recirculation.

Key to containment-failure modes:

α = CRSVE = containment rupture due to reactor vessel steam explosion

β = CL = containment leakage

γ = CR-B = containment rupture due to hydrogen burning

δ = CR-OP = containment rupture due to overpressurization

ϵ = CR-MT = containment rupture due to meltthrough

Figure 11-6. Event tree for a large LOCA in a PWR plant. An asterisk indicates no core melt.

to mitigate the consequences of the initiating event. An example would be the failure of the component-cooling-water system, which supplies cooling water to the seals of the reactor-coolant pumps. The failure of these seals may lead to a small LOCA. However, a failure of the component-cooling-water system causes a failure of the emergency core-cooling system (ECCS) because of the functional dependences between these two systems. The result is a LOCA with no ECCS available to mitigate the initiating event. Thus, the ESF cannot appear on the event tree as being available (Collins and Hudson, 1979).

In some risk studies, the analysts have preferred to develop a plant-level seismic fault tree to identify different core-melt and release sequences (Zion Probabilistic Safety Study--see Section 11.2.11.1).

11.2.6.3 Fault Trees

The major difference between earthquakes and internal events lies in the quantification of the fault trees. The frequencies of failure estimated for each component in a seismic fault tree are comprised of both the seismic fragility-related failure frequency and the random unavailability of the component. Each fault tree is expressed as a union of minimal cut sets. Calculation of the failure frequency for the system should consider the joint frequency distribution of the seismic responses and capacities of all components in the minimal cut set.

11.2.7 CONSEQUENCE ANALYSIS

A consequence analysis for seismic events differs from that for internal events in that some parameters of the consequence-analysis model may be influenced by the earthquake. For example, a large earthquake may disrupt the communications network and damage the roads that would be used for evacuation. It may also invalidate the consequence-modeling assumption that people will seek shelter in nearby buildings from external irradiation by gamma rays. In the presence of multiple hazards (i.e., earthquake and reactor accident), people may react differently than they would if faced with a reactor accident alone. For such reasons, the spatial distribution of population exposed to radiation effects in a seismically induced reactor accident is expected to be different from that for internal events. Similarly, there are some differences in the expected property damage for the two events. The consequence analyst should recognize these differences in building the consequence-analysis model for seismic events. The final output of the consequence analysis is a family of risk curves (Figure 11-7).

In recent seismic risk studies that included a consequence analysis, the consequence modeling has not been consistently different for seismic and internal events. This modeling approach was justified on the grounds that the large uncertainties assigned to the parameters of the consequence model are assumed to cover the differences between internal and seismic events.

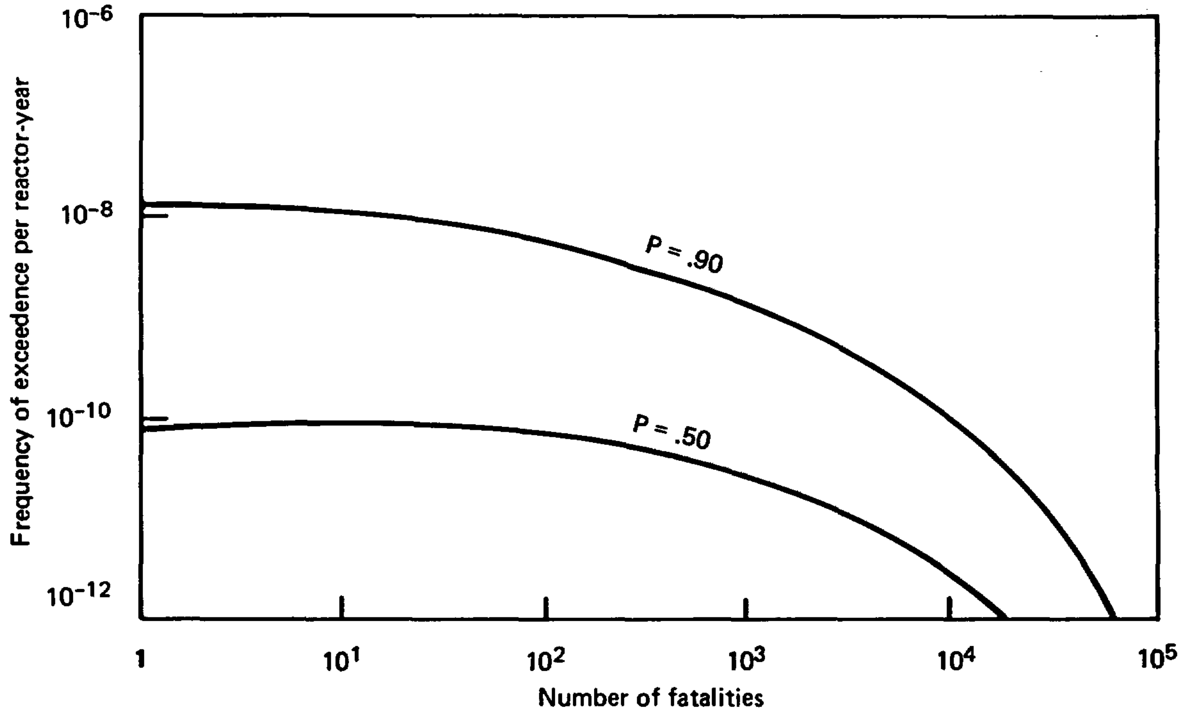


Figure 11-7. Seismic risk curves.

11.2.8 TREATMENT OF UNCERTAINTY

11.2.8.1 Sources of Uncertainty

Uncertainties in the analysis of seismic risk arise mainly from lack of data and analytical models. The sources of uncertainties are further grouped according to the stage of the analysis.

Seismic Hazard Analysis

The parameters of a model for seismic hazard analysis are separated into those that represent the inherent randomness of the seismic hazard and those whose values are uncertain. This distinction is also made for the sake of convenience in presenting the results of the hazard analysis and is based on the analyst's judgment. The first category contains the parameters whose values are estimated from empirical data. Examples are the activity rates of seismic sources, the mean attenuation relationship along with the dispersion, and relationships between intensity and acceleration, magnitude and rupture length, as well as intensity and magnitude with their respective dispersions in the data. The parameters whose values are uncertain include the geometric configuration of a seismic source, the value of the Gutenberg-Richter slope parameter b_0 for the source, the upper-bound magnitude or epicentral intensity for the source, and a cutoff value for the effective peak ground acceleration. The uncertainty in these parameters depends heavily on the specific site region. As such, no ordering of these parameters according to their contributions to the total uncertainty can be made.

Analysis of System and Structure Responses

In this portion of the seismic risk analysis, there are uncertainties in the seismic input and in the description of the dynamic behavior of the soil, structures, and subsystems (Johnson et al., 1981). In the definition of seismic input, uncertainties arise from using a limited number of parameters (e.g., peak ground acceleration) to describe the complex process of earthquake ground motion. Uncertainties are present in the definition of the ground-response spectra for a given peak free-field ground acceleration. Uncertainties in soil-structure interactions come from the idealization of the soil-structure system, the estimates of in-situ soil properties, and the details of the solution process. Uncertainties in structure responses result from variations in material properties (which affect structural frequencies, damping, etc.), variations in the details of construction, and the assumptions made in idealizing the structure. Uncertainties in piping-system responses arise from variations in material and geometric properties as well as modeling assumptions (e.g., linearity, gapless rigid supports, and the assumption that piping analysis can be decoupled from the structural analysis).

Component-Fragility Evaluation

The uncertainties in the component-fragility evaluation arise from an insufficient knowledge of material properties (e.g., strength and capacity for inelastic energy absorption), the definition of failure modes, the use of engineering judgment and generic data in lieu of complete plant-specific data, the lack of fragility test data for equipment, and the lack of data on the correlation between component capacities.

Plant-System and Accident-Sequence Analysis

The sources of uncertainty in the plant-system and accident-sequence analysis are the incomplete identification of all potential accident sequences, the lack of data on the physical interactions between components, and the modeling of dependences between component failures.

Consequence Analysis

In consequence analysis, the source of uncertainty specific to the seismic risk analysis is the lack of models for predicting the effects of large earthquakes on the parameters of consequence-analysis models (e.g., evacuation time, population distribution, and public response). Even if such models are available in other fields (e.g., lifeline earthquake engineering), the nuclear plant PRAs that have been conducted to date did not make explicit use of them.

11.2.8.2 Procedures for Uncertainty Analysis

At present, the quantification of uncertainties arising from different sources has to be done by a combination of limited analysis, sparse empirical data, and engineering judgment based on expert opinion. References that demonstrate how this quantification is accomplished include the Zion

Probabilistic Safety Study (Commonwealth Edison Company, 1981) and the reports of the Seismic Safety Margins Research Program (Smith et al., 1981; Johnson et al., 1981).

In the seismic risk studies performed so far, two essentially similar approaches to the propagation of uncertainties have been followed. One approach is to perform the risk analysis in two stages. In the first stage, the risk assessment is done with the best estimates of the parameters (about which there is uncertainty) of the seismic hazard analysis, response analysis, fragility evaluation, plant-system and accident-sequence analysis, and consequence analysis. This best-estimate analysis also assists in identifying the dominant accident sequences. Sensitivity studies with different parameter values are used to identify the significant parameters. In the second stage, a risk assessment of dominant sequences is repeated many times, each time with a different set of values for the significant parameters. These sets are sampled from the probability distributions of the parameters. By performing this two-stage analysis a sufficient number of times, one obtains the probability distributions for core-melt frequency, the frequency of each release category, and the frequency of exceeding various damage indices.

The other approach to the propagation of uncertainties is to assign discrete probability distributions (DPDs) to the parameters and then to use DPD arithmetic along with the Boolean expressions for the dominant accident sequences derived from fault trees to obtain a family of plant-level fragility curves for core melt and for each release category. Integration over the hazard-curve family yields probability distributions for core-melt frequency and the frequency of each release category. The same approach is extended into the consequence analysis to obtain a family of risk curves.

In performing the uncertainty propagation, it is important to maintain the "correlation" along the acceleration axis. A consistent way of doing this is as follows: The seismic hazard is represented by a set of hazard curves; associated with each curve is a probability (or "weight") p_i that reflects the analyst's degree of belief in the particular hypothesis. The entire seismic risk analysis is to be made conditional on one seismic hazard curve and is repeated for other curves. Similarly, component fragility is expressed by a set of fragility curves, each with an associated probability q_i. The integration over the entire range of acceleration values is performed for one fragility curve at a time. This ensures that the conditional frequency of failure does not decrease when the acceleration increases.

11.2.8.3 Available Information on Uncertainty Evaluation

In the Zion study (Commonwealth Edison Company, 1981), uncertainties in the parameters of the risk model were propagated throughout to obtain the probability distributions for core-melt frequency, the frequencies of various release categories, and the frequencies of exceeding various damage indices (early fatalities, thyroid cancers, etc.). The 10- to 90-percent probability range for the annual core-melt frequency is approximately 1×10^{-7} to 1×10^{-5}; for the annual occurrence frequency of release category 2R (i.e., delayed overpressure failure of containment without sprays), it is 2 x 10^{-7} to 2 x 10^{-5}.

It is observed that the uncertainty in the seismic hazard drives the total uncertainty in the seismic risk. The uncertainty in the seismic hazard stems mainly from differences in opinion between seismologists as to the upper-bound earthquake magnitude, the seismogenic source boundary, and the value of the Richter-Gutenberg slope (b). The many alternative hypotheses considered and the subjective probabilities assigned to them increase the uncertainty in the seismic hazard and thereby the uncertainty in the seismic risk. In contrast to the other portions of the seismic risk analysis (e.g., seismic fragility evaluation), the variables that introduce uncertainty in seismic hazard estimates are known, but the values of these variables are uncertain.

Some preliminary studies on uncertainties were done in phase I of the Seismic Safety Margins Research Program. A detailed investigation of the uncertainties in seismic risk analysis coupled with sensitivity studies is being undertaken in phase II.

11.2.9 FINAL RESULTS OF A SEISMIC RISK ANALYSIS

The final results of a seismic risk analysis may take several forms, depending on the site and the specific objectives of the PRA. If the site is in a low-seismicity region, it may be sufficient to calculate the frequency of an earthquake-induced core melt for comparing with the frequency of core melt from internal events and other external events. The final results of a seismic risk analysis can then be presented as shown in Figure 11-1 for a hypothetical plant. In this figure, the median annual frequency of an earthquake-induced core melt is 1 x 10^{-6}. The 5- to 95-percent probability ("confidence") interval for the annual core-melt frequency is 1 x 10^{-7} to 1 x 10^{-5}.

If the annual frequency of an earthquake-induced core melt is significant in comparison with other internal and external events, the objective of the PRA study may be extended to estimate the radiological risk from seismic events. An intermediate result of the seismic risk analysis can then be presented as a probability density function for the frequency of each release category (Figure 11-2). Again, these frequencies can be compared with the release frequencies from other internal and external events to judge the seismic event contribution to plant risk. Finally, the frequencies of exceeding different damage levels (e.g., number of early fatalities) can be presented at different probability levels (Figure 11-7).

If the contribution from the seismic risk dominates the total plant risk and the latter is considered to be high, it is necessary to review the components and systems in the release sequences that had high frequencies of occurrence. It may be necessary to reevaluate the seismic hazard curve, component responses, and component fragilities by using more-detailed models and by gathering additional data. Such a review may also uncover weak links in the methods of the seismic design (Smith et al., 1980).

In recent years, many PRA studies have been initiated for operating nuclear power plants as well as those under construction. Since these studies are still in their preliminary stages and not all results have been published, the relative contribution of seismic events to plant risk and the relative significance of each component and system in the seismic safety of the plant are not yet established. Specific guidelines for the appropriate level of effort in the seismic risk analysis cannot therefore be given at this time.

11.2.10 REQUIREMENTS FOR SEISMIC RISK ANALYSES

A review of the seismic risk studies performed so far indicates that they vary in completeness and sophistication, mostly because of differences in their objectives and scopes. As already mentioned, the analysis of seismic risk has not reached a stage where definitive guidelines as to the level of detail can be provided. However, it may be advisable to outline a few general requirements in order to promote uniformity and consistency in PRA studies. For a seismic risk study to be acceptable, it is not sufficient to just meet these requirements: the study must be conducted by a qualified team and be thoroughly reviewed by peers.

The requirements are as follows:

1. The analysis should consider all the plant systems and components whose failures might contribute importantly to the frequency of release.

2. The analysis should include all significant variables contributing to the seismic hazard, to the responses of structures and equipment, to the fragilities of components and systems, to the release frequencies, and to the plant risk.

3. At each stage of the analysis, the analyst should not only make a best estimate of each variable but also record the uncertainty in the estimate. The seismic hazard analysis should reflect variations in professional opinions regarding the values of different variables (e.g., upper-bound earthquake magnitude, seismic source boundaries, and the value of b_0 in Equation 11-6). The uncertainties in different variables should be consistently propagated so that the confidence in the output (e.g., core-melt frequency and risk estimates) can be quantified.

4. The risk-analysis method should not be constrained by the plant licensing criteria. For example, the inelastic capacities of structures and equipment should be estimated, although the plant design criteria may require that the structures and equipment be within yield levels under the safe-shutdown earthquake. In the seismic hazard analysis, the entire range of earthquake levels should be studied, even though the plant may have been designed for a conservatively selected safe-shutdown earthquake.

5. Since seismic events have the potential to affect a number of components, it is essential that any correlation between the failures of different components be properly handled. Correlation arises both in the responses of different components to a single earthquake and a common structural model, and in component capacities (because of a common manufacturer or similar mounting). The correlation due to a common earthquake level is automatically handled by integration in Equation 10-1, where the product of the frequency of occurrence of any earthquake level and the conditional frequency of failure for a sequence of components given the hazard is integrated over the entire range of hazard intensity.

6. The possibility of equipment failure from nonseismic causes (e.g., random failures) during a seismic event should be recognized. For example, the ceramic insulators on offsite-power transformers have very high frequencies of failure at moderate ground accelerations (0.20g to 0.30g). Given such a failure and the resulting loss of offsite power, the unavailability of emergency diesel generators in the time required to repair or replace the insulators should be considered. It may be greater than the unavailability attributable to the seismic fragility of diesel generators at low to moderate earthquake levels (0.20g to 0.40g).

11.2.11 CURRENT METHODS

Two methods are currently available for estimating seismic risks. They generally fulfill all of the requirements outlined in Section 11.2.10. The major difference between them is the level of detail in the seismic response analysis and the plant-system and accident-sequence analysis. The first method was developed and applied in the Oyster Creek PRA study (Garrick and Kaplan, 1980; Kennedy et al., 1980). It has since been improved and applied to estimate seismic risks for the Zion plant (Commonwealth Edison Company, 1981) and the Indian Point plant (PASNY, 1982). Called here "the Zion method" for short, it is now being used in estimating seismic risks for the La Salle, Oconee, Browns Ferry, Midland, and Pilgrim plants. The second method was developed in an NRC-funded research program at the Lawrence Livermore National Laboratory--the Seismic Safety Margins Research Program (Smith et al., 1981); it is called "the SSMRP method" in the discussion that follows. By coincidence, the Zion Nuclear Generating Station was selected as a reference plant for the SSMRP study.

The Zion method relies heavily on the use of engineering judgment to supplement sparse data and limited analysis, whereas the SSMRP method emphasizes extensive component and system modeling as well as a detailed seismic response analysis. Engineering judgment is, of course, used in the SSMRP method in estimating the seismic hazard, deriving component fragilities, and performing the plant-system analysis. For a routine PRA study of a nuclear power plant, the Zion method offers a procedure that takes into account all the important features of the seismic risk and involves relatively less effort. However, the risk estimates derived by this procedure may have larger variabilities associated with them. If the Zion method shows that the seismic risk contribution dominates the total plant risk or that a particular plant safety system is a dominant risk contributor, a more refined estimate of the seismic risk can be obtained by following the SSMRP method.

11.2.11.1 The Zion Method

Seismic Hazard Analysis

The procedure for the seismic hazard analysis is essentially as outlined in Section 11.2.3. In the Zion study, the seismic hazard model was based on the tectonic provinces described in the work of TERA (1979), in which a number of nationally recognized seismicity experts made judgments of the seismicity in various regions of the United States. The earthquake data available in Modified Mercalli intensity units were converted to the body-wave magnitude m_b by means of Equation 11-12. The attenuation relationship between m_b and the sustained maximum ground acceleration a_s as given in Equation 11-13 was used. The uncertainty in the maximum m_b that the seismic sources are capable of producing was accounted for by assigning probabilities of .28, .44, and .28 to the m_b values of 5.6, 5.8, and 6.0, respectively. Similarly, three different tectonic province assumptions were made; probabilities of .5, .3, and .2 were assigned to these hypotheses (McGuire, 1981).

In accordance with the work of Kennedy (1981), the effective peak ground acceleration was expressed in terms of the sustained maximum ground acceleration, and the maximum cutoff values established for this parameter were 0.44g to 0.65g. Some sensitivity analyses have revealed, however, that the risk estimates are not too sensitive to the maximum cutoff values.

Analysis of Plant-System and Structure Responses

In the Zion method, structural and equipment fragilities are expressed in terms of a ground-motion parameter (e.g., effective peak ground acceleration). A response factor of safety is derived from a linear dynamic analysis of the structure or equipment. In most cases, the results of response analyses performed for the design-earthquake levels (e.g., operating-basis and safe-shutdown earthquakes) and ground-response spectra can be used to estimate the response factor of safety. As already mentioned, this factor of safety depends on the safety factors involved in the selection of ground-response spectra, the procedure used to include the effects of soil-structure interactions, the selection of damping levels, the modeling of structures and piping, and the method of analysis. The safety factors are treated as random variables, and their statistical parameters, such as the median and the logarithmic standard deviation, are estimated by using available data and engineering judgment.

Fragility Evaluation

Section 11.2.5.3 describes the development of component fragilities by the Zion method. Examples of the use of this method for deriving component fragilities can be seen in the Zion study (Commonwealth Edison Company, 1981) and in a recent report by Ravindra (1982).

Plant-System and Accident-Sequence Analysis

The plant-system and accident-sequence analysis used in the Zion method can be summarized as follows:

1. For each initiating event, the analyst constructs fault trees reflecting (a) failures that could initiate an accident sequence and (b) failures of key system components or structures that could mitigate the core-melt sequence.

2. The fragility of each such component (initiators and mitigators) is estimated.

3. Fault trees are used to develop Boolean expressions for core-melt sequences that lead to each of the various plant-state frequencies.

4. Considering possible core-melt sequences and containment mitigating systems (e.g., fan coolers, containment sprays, and containment), Boolean expressions are developed for each release category.

The plant model is described through a seismic fault tree like the one shown in Figure 11-8 for a PWR plant. According to this fault tree, an earthquake-induced core melt occurs if any of the initiating events occurs together with a loss of safety injection or loss of cooling; that is,

$$M_S = M_1 \vee M_2 \vee M_3 \tag{11-28}$$

where the Boolean-algebra symbol $\vee$ means "or"; in the sequel, the symbol $\wedge$ is used to signify "and." For each of these events, M_1, M_2, and M_3, fault trees are constructed in terms of the primary component failures. Figure 11-9 shows a typical fault tree for a small LOCA with a loss of safety injection or cooling.

The termination of a fault tree at any basic component failure level depends on the fragility of the component and on the maximum ground acceleration possible at the site. All components that have a slight chance of failure (e.g., 5 percent) at the upper-bound effective peak ground acceleration are included in the fault trees. These components are identified by reviewing the fragility descriptions developed earlier.

The frequency of core melt is calculated by combining the plant-level fragility with the seismic hazard curves. This is done by first translating the seismic fault trees into a Boolean expression. For example, the Boolean expression for an earthquake-induced core melt is

$$M_S = 1 \vee 2 \vee 5 \vee 6 \vee 7 \vee 4 \vee [(9 \vee 10 \vee 8) \wedge 3] \tag{11-29}$$

The components denoted by numbers in circles are listed in Table 11-1 generally in order of increasing capacity. Plant-level fragility curves are obtained by aggregating the fragilities of individual components according to Equation 11-29 and using DPD arithmetic (Kaplan, 1981). An example of the

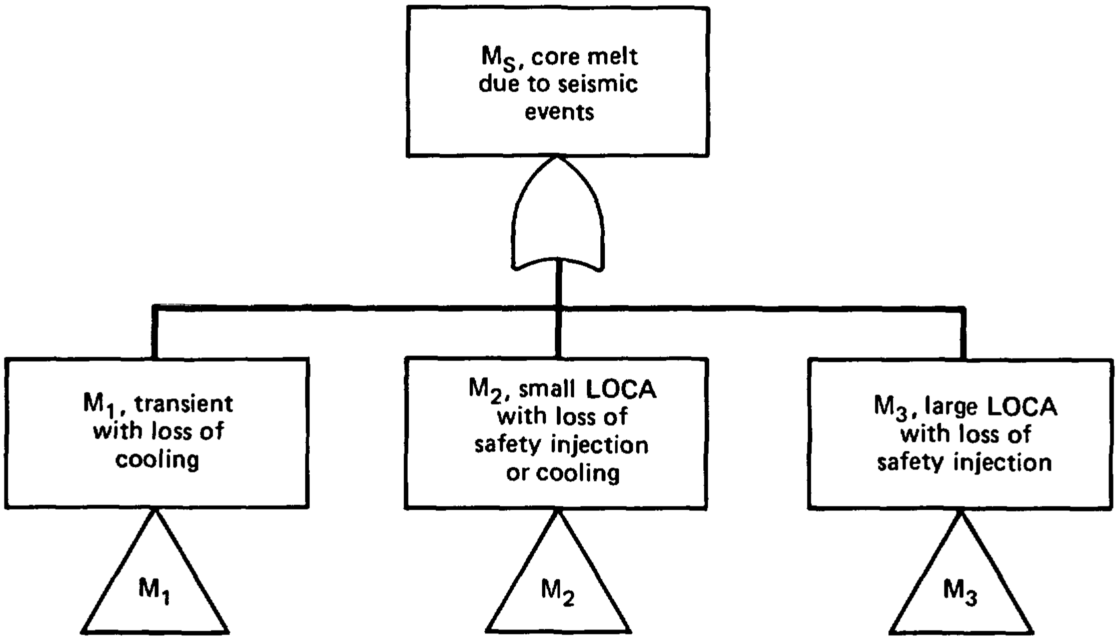


Figure 11-8. Seismic fault tree for a PWR plant.

plant-level fragilities is shown in Figure 11-10. For purposes of comparison, a family of fragility curves for a particular component is also plotted in Figure 11-10. We can observe a shift in the fragility curves from the component level to the plant level.

This shift to the left in the plant-level fragility illustrates an important feature of the seismic risk problem. Since an earthquake can simultaneously affect a number of redundant components, the plant-level fragility (i.e., the conditional frequency of failure given an acceleration value) is higher than the fragility of any component.

The core-melt frequency f_c is calculated as follows:

$$f_c = \sum_i h(a_i)\, f_S(a_i) \qquad (11\text{-}30)$$

where $f_S(a_i)$ is the occurrence frequency of a system failure that leads to a core melt for effective peak ground accelerations of less than or equal to a_i and $h(a_i)$ is the annual frequency of occurrence of earthquakes with an effective peak ground acceleration between a_i and $(a_i + \Delta a)$. The summation is carried over the entire range of accelerations.

The seismic hazard and the plant-level fragility are each represented by a family of curves plotted for different nonexceedence probabilities. Equation 11-29 and DPD arithmetic are then used for a probabilistic multiplication of these curves to obtain the probability distribution for the core-melt frequency (Figure 11-1).

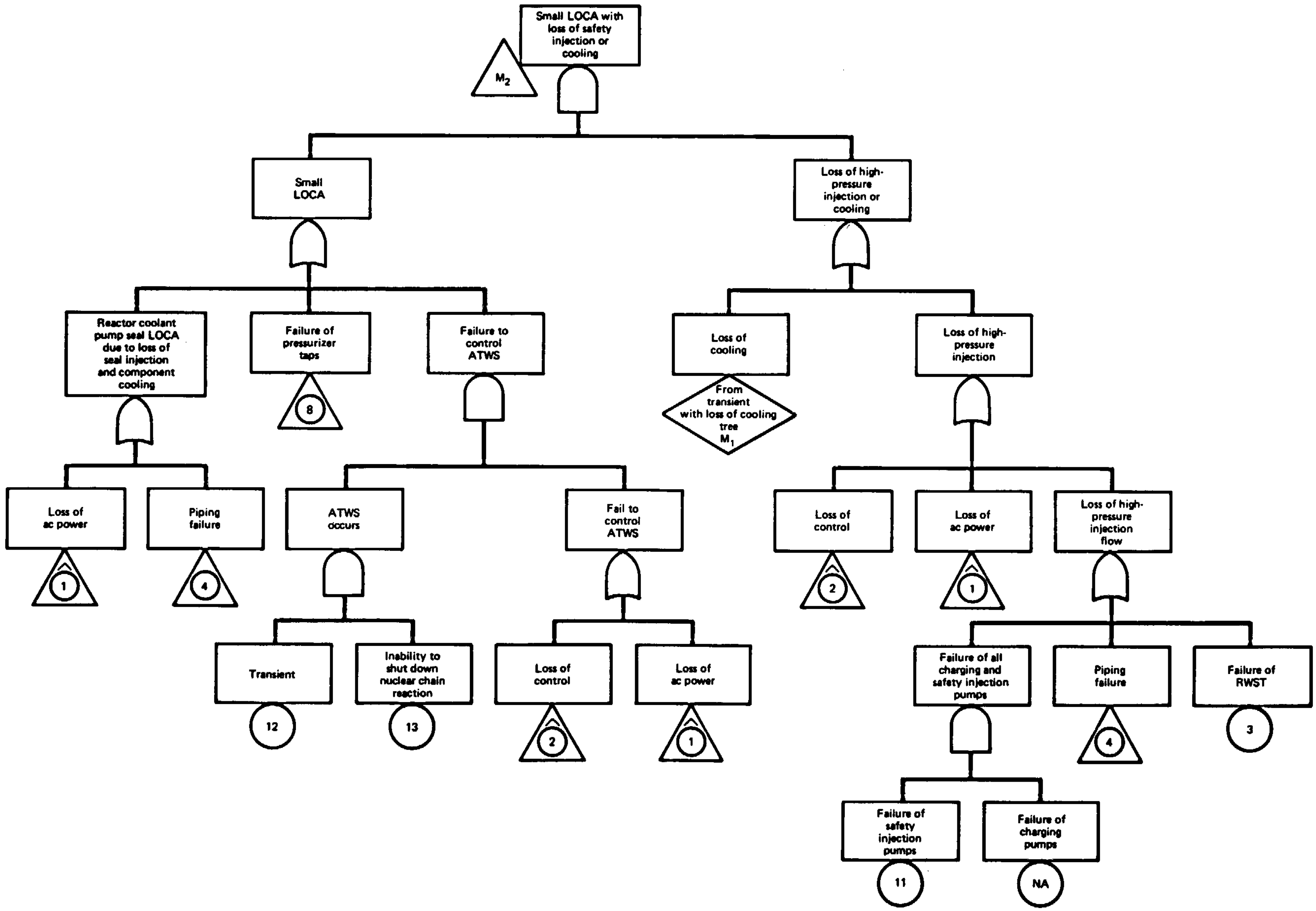


Figure 11-9. Fault tree for a small LOCA with loss of safety injection or cooling in a PWR plant. The numbers in circles correspond to the primary events of Table 11-1. The label "NA" (not applicable) means that this failure is not induced by the range of possible seismic events.

The frequencies of various release categories are based on the type of core melt (i.e., plant state) and the containment state (failure or no failure). The plant state would be dependent on the top event of the fault tree (i.e., M_1, M_2, or M_3) and on the functioning of the containment fan coolers and containment sprays. Boolean equations are developed for different plant states. Each plant state combined with the containment state is assigned to a particular release category. Therefore, a Boolean equation for each release category is derived to express the logical relationships between component failures. Using the Boolean equation along with component-fragility families, a fragility family for each release category is derived. By integrating this family of fragility curves over the family of seismic hazard curves, a probability distribution is obtained for the frequency of the release category (see Figure 11-2).

Table 11-1. List of critical structures and equipment in the seismic fault tree for a typical PWR[a]

Number[b]	Structure or equipment
1	Service-water pumps
2	Auxiliary building--failure of concrete shear wall
3	Refueling water storage tank
4	Interconnecting piping/soil failure beneath reactor building
5	Collapse of pump enclosure roof in cribhouse
6	125-volt dc batteries and racks
7	Service-water system, 48-inch buried pipe
8	Collapse of pressurizer-enclosure roof
9	Condensate storage rack
10	20-inch piping, condensate storage tank
11	Safety injection pumps
12	Offsite-power ceramic insulators
13	Core geometry

[a]See Figure 11-9 and Equation 11-29.
[b]Shown inside circles in Equation 11-29 and Figure 11-9.

A computer code called SEIS has been developed (Kaplan, 1981) to perform the probabilistic calculation of core-melt and release frequencies. The seismic hazard curves and the component-fragility families are the inputs to this code.

Consequence Analysis

In the Zion probabilistic study (Commonwealth Edison Company, 1981), the consequence model developed for internal events was employed for analyzing the

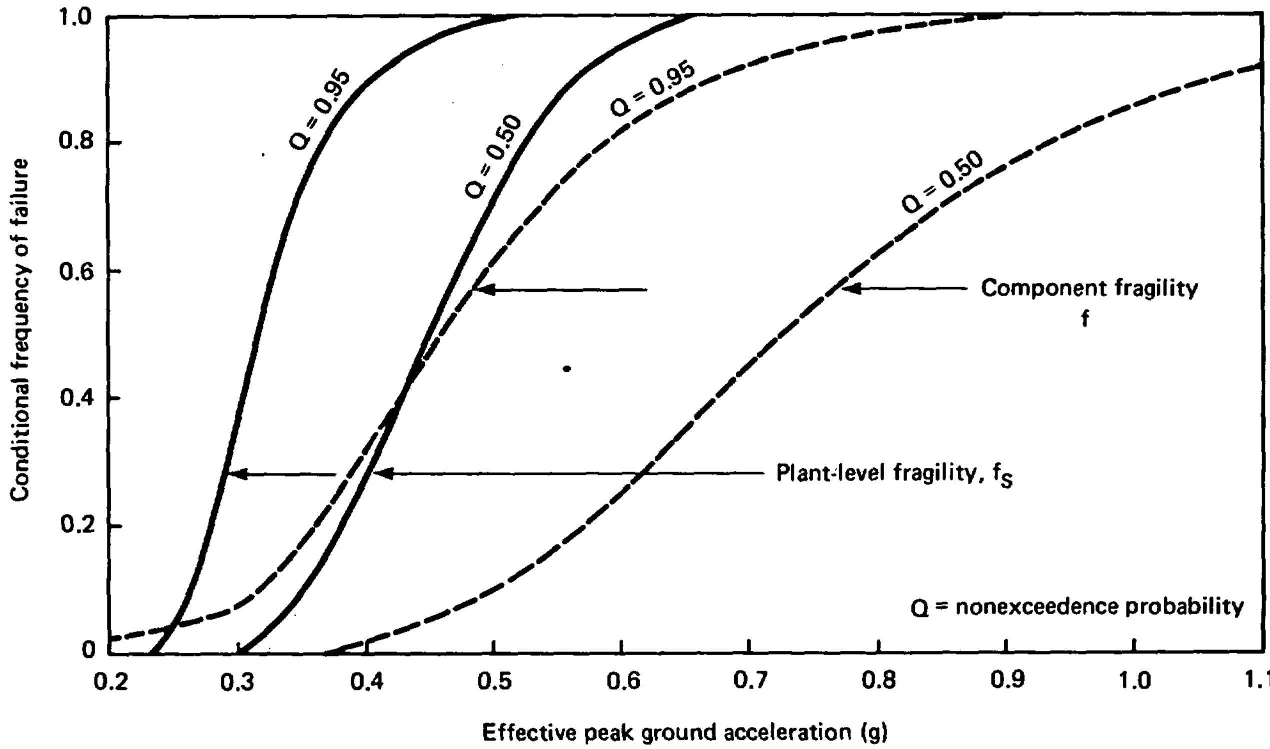


Figure 11-10. Component and plant-level fragility curves.

consequences of seismic events. The reasoning was that the large uncertainties assigned to the parameters of the consequence model would cover the variations due to seismic events.

11.2.11.2 The SSMRP Method

Seismic Hazard Analysis

The procedure for the seismic hazard analysis is essentially as outlined in Section 11.2.3. In the first phase of SSMRP, the peak ground acceleration was used as the hazard parameter. Historical data and expert opinion were used in calculating the annual frequencies of exceeding different values of the peak ground acceleration.

Analysis of Plant-System and Structure Responses

In the SSMRP method, structural and equipment fragilities are expressed in terms of local response parameters, such as stress, moment, and spectral acceleration. Therefore, given an earthquake, the conditional failure frequency of a structure or equipment is obtained by a convolution of the frequency distribution of the response for that ground acceleration and the frequency distribution of the resistance of the structure or equipment.

A major emphasis of the SSMRP method lies in the computation of structural and equipment responses. Some of the distinguishing features of the SSMRP response-analysis method as developed and applied in phase I of the program are summarized below.

As mentioned above, the seismic hazard parameter is the peak ground acceleration. Variability in the ground motion is incorporated into the analysis by simulating a set of time histories consistent with the hazard curve; each time history is developed to simulate a particular spectral shape.

A detailed analysis of soil-structure interactions is made, using the substructure approach. For structures, a detailed finite-element analysis is performed. Subsystem responses are determined by using a multisupport time-history analysis. The subsystems modeled in phase I of the program consisted of a number of valves, nozzles, and pumps as well as critical piping nodes.

Uncertainties in the input parameters (e.g., soil shear modulus and damping, and structures and subsystem frequencies and damping) are included by using a Latin-hypercube technique to sample different parameter values for each earthquake time-history simulation (Iman et al., 1980).

By such detailed modeling and analysis, phase I of the SSMRP study was able to derive the peak responses of structural elements, equipment, piping nodes, etc., and the correlation between them. This extensive response analysis was needed because the phase I study was a first attempt at quantifying the probabilistic response with all the input variables included and because the reference plant was not designed to present-day criteria and standards. Furthermore, the objective of the program was to establish the relative importance of various methods used in current seismic design practice.

Phase II of the SSMRP will develop methods for estimating the ratio of the actual response to the design response for structures and equipment in nuclear power plants designed according to the procedures of the NRC Standard Review Plan (Bumpus et al., 1980). This should circumvent the need for a detailed response analysis and should facilitate an optimal use of plant-design information. Phase II will also concentrate on estimating the sensitivity of response to different input parameters in order to improve the efficiency of response computations and will investigate the significance of assuming perfect dependence or perfect independence between component responses in lieu of a detailed correlation analysis.

Fragility Evaluation

As discussed before, in the SSMRP method the fragility of a component is anchored to local response parameters (e.g., moment, stress, and spectral acceleration). Therefore, the fragility description for a component includes only the variability of its seismic capacity. In this the SSMRP method differs from the Zion method, which expresses the fragility of a component as a function of both seismic response and capacity. Apart from this, the procedure for developing component-fragility curves is the same as that described in Section 11.2.5.2. The fragilities of structures are based on the peak ground acceleration at which inelastic structural deformations would interfere with the operation of safety-related equipment housed in the structure. This failure acceleration is computed as the SSE acceleration times a factor of safety that accounts for original conservative estimates of material strength and conservative design analysis. The factor of safety also includes a ductility factor that allows nonlinear failure criteria to be related to the linear responses calculated as described in Section 11.2.4.

Details of how the fragility curves for structural elements were established in phase I of the SSMRP study are given by Wesley and Hashimoto (1980). Equipment and piping fragility was described by a random variable C that represents the seismic capacity expressed in terms of force, moment, or spectral acceleration. As before, the random variable C was modeled as

$$C = F_C A_{SSE} \tag{11-31}$$

where A_{SSE} is the magnitude of the fragility (local response) parameter specified for the safe-shutdown earthquake and F_C is an equipment-capacity factor that accounts for both strength and ductility. The development of fragilities for equipment and piping has been described by Campbell et al. (1981).

In phase I, randomness and uncertainty were not treated separately; instead a composite fragility curve was used for each component. The total variability was approximated as

$$\beta_C = \left(\beta_{C,R}^2 + \beta_{C,U}^2\right)^{1/2} \tag{11-32}$$

Correlations among component fragilities (owing to the same manufacturer or identical mounting), if specified by the analyst, can be consistently handled by the SSMRP computation scheme.

Plant-System and Accident-Sequence Analysis

The SSMRP method consists of identifying initiating events, developing event and fault trees, and finding the dominant accident sequences for various release categories. In phase I of this program, the occurrence frequencies of initiating events were calculated for various peak ground accelerations. An event tree was constructed for each initiating event, and from these event trees 148 core-melt sequences were identified. Each core-melt sequence was followed by a containment event tree that established the release sequence. Figure 11-6 shows an event tree for a large LOCA in a PWR (Smith et al., 1981); it contains 23 core-melt sequences, and each sequence can lead to a release through the potential containment-failure modes designated α, β, γ, δ, and ε.

A fault tree was used to evaluate the occurrence frequency of system failure at each branch of an event tree. Fault trees were developed for the auxiliary feedwater system, the service-water system, the emergency core-cooling system (comprising the safety injection system), the residual-heat-removal system, the charging system and the accumulators, and the electric power system. Each fault tree had several hundreds of components and was represented by a union of cut sets. The frequencies of primary events in a fault tree, and hence the frequency of system failure, depend on component-failure frequencies. In calculating the component-failure frequencies, both random failures and earthquake-induced failures were taken into account. The frequencies of earthquake-induced failures were calculated by using a multivariate peak-response distribution developed by the SEISIM code from the output of the SMACS code and the component fragility developed as described in Section 11.2.5.

The computer code SEISIM (Seismic Evaluation of Important Safety Improvement Measures) was developed to compute the occurrence frequencies of structural failures, component failures, system failures, and radionuclide releases (Hudson and Collins, 1979). It uses as input the seismic hazard curves, the component-fragility curves, and the structure and equipment responses calculated by the SMACS code. Event and fault trees are represented by Boolean expressions. A unique feature of SEISIM is the consistent treatment of the correlation between component responses to a given earthquake and between component capacities.

The SEISIM code computes the conditional occurrence frequency of a single radionuclide-release sequence (for a given earthquake-acceleration range) as the product of four occurrence frequencies: (1) the frequency of an earthquake producing the given input ground-motion level, taken from the seismic hazard curve; (2) the frequency of the necessary initiating event, given the input motion level; (3) the frequency of the core-melt accident sequence, given the input motion level and the initiating event; and (4) the frequency of the containment-failure mode producing the specific radionuclide release, given the input motion level, the initiating event, and the accident sequence. The SEISIM code accumulates these conditional frequencies of release sequences into each of the seven radionuclide-release categories identified in the Reactor Safety Study (USNRC, 1975). Finally, the unconditional frequency of release for a given release category is obtained by integrating over all acceleration ranges. SEISIM is also structured to perform sensitivity analyses to determine which components or systems dominate in the failure- and release-frequency computations.

In the first phase of the SSMRP, the uncertainties identified in the seismic hazard analysis and in the development of component fragilities were not completely propagated. Extensive sensitivity studies and uncertainty propagation are planned for the second phase. The two-stage analysis described in Section 11.2.8 and illustrated by Collins and Hudson (1981) may be used for this purpose.

Consequence Analysis

In the first phase of the Seismic Safety Margins Research Program, the frequencies of radionuclide releases were the end products. An analysis of consequences was not performed. If the analyst elects to use the SSMRP method, he may follow the consequence-analysis method described in Section 11.2.7.

11.2.12 INFORMATION AND PHYSICAL REQUIREMENTS

11.2.12.1 Information Requirements

The information needed to perform a seismic risk analysis for a nuclear power plant consists of plant-design details as well as generic information.

Plant-Design Details

The needed information includes the following:

1. Description of plant systems, including the location of structures and components, and sizes of structural members; a set of general arrangement, structural, piping, electrical, and equipment drawings.

2. Design criteria, applicable codes, and applicable standards.

3. Safety analysis report, especially the chapter on the geologic and seismic characteristics of the region in which the site of the plant is located.

4. Material-strength test reports (i.e., concrete-cylinder test data and steel-mill certificates).

5. Design reports for plant-specific equipment, the nuclear steam supply system, and engineered safety features.

6. Specifications for the seismic design of equipment.

7. Reports on qualification and preservice tests as well as periodic inservice inspections.

8. Stress reports, including structural and subsystem models for the seismic response analysis.

9. Seismic Qualification Review Team (SQRT) reports for equipment, if available.

Generic Information

The list of needed generic information includes the following data and reports:

1. High seismic zone qualification reports for identical and similar equipment.

2. Seismic capacities of similar valves. (In the probabilistic risk assessments performed to date, plant-specific information on valves

was not available, and therefore it was necessary to use generic information.)

3. Shock-test reports from the U.S. Army Corps of Engineers or other sources.

4. Topical reports.

5. Reports of published PRA studies.

11.2.12.2 Personnel and Schedule

In addition to system analysts and reliability engineers, the PRA study team should include a qualified seismologist and engineers experienced in seismic hazard analysis, seismic structural and subsystem analyses, structural and mechanical design, and seismic qualification testing. Since the hazard analysis and seismic fragility evaluation call for engineering judgment to supplement the results of simplified analyses and sparse test data, the PRA study team may benefit from seeking advice from outside experts in these fields.

Seismic hazard analysis and fragility evaluation may start at the beginning of a PRA project; however, plant-system and accident-sequence analyses and the seismic risk assessment can be efficiently done after the event and fault trees for internal events are developed.

Computer requirements would not be different from those for the internal events. If a more detailed response analysis and release-frequency analysis are attempted, nonproprietary computer codes like SMACS and SEISIM can be used.

11.2.13 PROCEDURES

The recommended task-by-task procedure for performing the seismic risk analysis of a nuclear power plant is given below.

Task 1: Collection of Information

The collection of information on plant design, regional seismology, test reports, etc., as described in Section 11.2.12.1, forms the starting point for the seismic risk analysis.

Task 2: Establishment of Objectives and Scope

The objectives of the seismic risk analysis need to be established; it could be a part of the routine PRA done for the plant, or the seismic risk analysis may have been necessitated by a situation not envisioned during plant design (e.g., the discovery of a potential fault close by). The scope of the analysis and the refinement of the analytical model will depend on the objectives of the seismic risk analysis. A more refined procedure may be used if the seismic safety of the plant is to be proved in light of new information (e.g., a potential fault).

The project team should agree on a description of the ground-motion parameter (e.g., effective peak ground acceleration and instrumental peak ground acceleration). This will establish proper communication between the seismologist, the structural engineer, and the systems analyst. Similarly, different failure modes for structures and equipment should be defined and quantified.

A format for reporting the results of intermediate tasks is to be established. A possible format is the probability of frequency. The seismic hazard, the component fragility, and the release frequency can all be presented in this format to ensure consistency between different tasks.

The method chosen for the seismic risk analysis should meet the requirements outlined in Section 11.2.10.

Task 3: Plant Familiarization

Before a detailed analysis can begin, it is necessary for the PRA team to become familiar with the design, operation, and maintenance of the plant. Plant-design criteria, stress reports, design and as-built drawings, qualification procedures for equipment, the functions of various plant systems, and consequences of failures should be reviewed to aid in identifying initiating events and in constructing the models for plant-system and accident-sequence analyses and for the consequence analysis. A walk-through inspection of the plant is essential to identify the status of component supports (equipment and piping) and any visible deviations from the as-built drawings.

Task 4: Seismic Hazard Analysis

The seismology and past earthquake history of the site region should be reviewed. The information documented in the safety analysis report is a good starting point. A seismic hazard model identifying all seismic sources in the site region should be developed. The parameters of this model, such as source boundaries, activity rates, recurrence relationships, the upper-bound magnitude or the epicentral intensity of each source, attenuation relationships, and the correlation between intensity and ground acceleration, should be established on the basis of site-specific data, applicable regional data, and the expert opinion expressed in professional papers and reports.

Alternative hypotheses reflecting professional uncertainty on the values of such significant variables as source boundaries, the upper-bound earthquake magnitude, and the Richter-Gutenberg slope (b) should be postulated. A probability (or weight) should be assigned to each hypothesis.

Task 5: Analysis of Plant-System and Structure Responses

If a method like the Zion method is chosen, no new response analyses for structures and equipment need be performed; the responses for earthquakes different from the safe-shutdown earthquake can be obtained by extrapolation from the design responses. When a method like the SSMRP method is used, it may be necessary to develop detailed analytical models of structural systems and piping subsystems if no such models exist or the design analysis models are not adequate; the response analysis is performed with a computer code like SMACS.

Task 6: Fragility Evaluation

Depending on the risk-analysis method that is chosen, component fragilities will be developed as a function of a global ground-motion parameter or a local response parameter. A list of safety-related structures, piping, and mechanical and electrical equipment should be provided by the systems analyst to the structural engineer assigned to this task. By reviewing the plant design bases, the structural engineer will estimate the median inelastic safety factor implied by the component design over the SSE acceleration (response) for the particular mode of failure. He would also express the variability in the safety factor by the values of β_R and β_U, inherent randomness and uncertainty. The fragility curve is therefore represented by the median ground acceleration (or local response) capacity and the values of β_R and β_U.

Task 7: Plant-System and Accident-Sequence Analysis

This analysis begins with the identification of the earthquake-related initiating events, such as a large loss-of-coolant accident, a small loss-of-coolant accident, and transients. Event trees and/or fault trees showing core-melt and radionuclide-release sequences are developed for each initiating event. If event trees are used, the failure at each branch of the event tree is represented by generating a fault tree. The core-melt frequency and the frequencies of release categories are calculated from the seismic hazard estimates, the component fragilities, and the event and fault trees. Dependences between component failures should be properly accounted for in the analysis.

Task 8: Consequence Analysis

The consequence analysis specific to the seismic event is performed with a consequence model that may be a modification of the model used to analyze the consequences of internal events to reflect the effects of earthquakes on the evacuation of people, public response, etc.

Task 9: Development and Display of Results

The results of a seismic risk analysis consist of seismic hazard curves, component fragilities, probability distributions for the occurrence frequency of earthquake-induced core-melt accidents and for the occurrence frequencies of various radionuclide-release categories, and risk curves. Other useful results include failure frequencies for structures, systems, and equipment and the accident sequences that dominate the seismic risk.

11.2.14 METHODS OF DOCUMENTATION

The chapter of the PRA report that discusses the seismic risk analysis should include the sections described below.

Description of the Site Region

The geographic location of the plant as well as the seismic and geologic characteristics of the site and the region surrounding the site should be described.

Analytical Method

The reasons for choosing a particular risk-analysis method should be discussed, demonstrating that the chosen method meets the requirements of the seismic risk analysis (Section 11.2.10).

Seismic Hazard Analysis

The report should describe the hazard model; the seismic sources, their activity rates, and upper-bound magnitude or epicentral intensity; recurrence relationships and available intensity or magnitude data; the attenuation relationship selected for the analysis; and the correlation between intensity and magnitude and/or acceleration if used. Uncertainties in these parameters of the hazard model should be discussed in detail, along with the methods used to quantify them (see Section 11.2.8 for a discussion of the treatment of uncertainties). Sources of data, professional papers, and opinion surveys should be included.

The final result of the hazard analysis should be presented as a family of annual exceedence-frequency curves plotted against the values of a ground-motion parameter. The choice of the ground-motion parameter should be substantiated.

Analysis of Plant-System and Structure Responses

If the structural and subsystem responses for earthquake-acceleration levels higher than those of the safe-shutdown earthquake were obtained by a linear extrapolation, a brief description of the design analysis should be given. If a more detailed structural and subsystem analysis was performed, a description of the analytical models used for the seismic input and for soil, structures, and subsystems should be given. The input parameter values selected in this analysis (e.g., soil properties, structural damping, and ductility) should be reported. Variability in the response as a result of uncertainties in the input parameters and in the analytical models should be quantified.

Fragility Evaluation

The report should describe the failure modes of the structures and the equipment for which the fragility curves were developed. The discussion should include both the sources of data and the methods used in developing fragilities for structures and equipment. A tabulation of safety-related structures and equipment as well as their fragility parameters should be provided.

Plant-System and Accident-Sequence Analysis

Starting with a discussion of the initiating events selected for analysis, this section should describe how their frequencies of occurrence were estimated, describe the event and fault trees for each initiating event, and list all identified accident sequences. It should also describe how the core-melt frequency and the frequencies of release categories were calculated and document how dependences between component failures were accounted for in the analysis.

Consequence Analysis

The parameters of the consequence-analysis model that are different from those traditionally used for internal events (see Chapter 9) should be described, substantiating the values chosen.

Final Results

The results of the seismic risk analysis should include the seismic hazard curves, families of component fragilities, and modifications to the event and fault trees of internal events. If the seismic risk analysis is separately carried further, the probability distribution for the occurrence frequency of an earthquake-induced core-melt accident and probability density functions for the occurrence frequencies of various radionuclide-release categories should be presented. Finally, the seismic risk curves should be presented for selected damage indices, examples being early fatalities, latent-cancer fatalities, and property damage. The accident (release) sequences and the systems and/or components that are the dominant contributors to public risk should be identified.

11.2.15 DISPLAY OF FINAL RESULTS

The results of a seismic risk analysis consist of the following:

1. Seismic hazard curves (Figure 11-4).
2. Component fragilities (Figure 11-5).
3. Probability distribution for the occurrence frequency of earthquake-induced core-melt accidents (Figure 11-1).
4. Probability density functions for the occurrence frequency of radionuclide-release categories attributed to earthquakes (Figure 11-2).
5. Seismic risk curves (Figure 11-7).

11.3 RISK ANALYSIS OF FIRES

11.3.1 INTRODUCTION

The early applications of risk analysis to nuclear power plants, including that presented in the draft report of the Reactor Safety Study (RSS), did not include a quantitative assessment of accidents initiated by major fires. The reason for this omission was twofold: (1) it was judged that fires were not likely to be dominant contributors to risk (RSS final report--USNRC, 1975) and (2) the state of the art in risk analysis had not yet developed an approach to covering fires. The importance of fire as a potential initiator of multiple-system failures took on a new perspective after the cable-tray fire at Browns Ferry in 1975. Although various experts have disagreed as to how close that fire came to an accident resulting in core damage and a major release of radioactive material, it is clear that its impact was extensive when measured in terms of the failure of redundant and diverse safety-related systems. It is not surprising, therefore, that risk analyses performed after the Browns Ferry fire have tended to include fires in the quantification of risk.

One of the first attempts at numerically estimating the risks due to fires appeared in the final report of the Reactor Safety Study, published later in the same year (1975) as the Browns Ferry fire. An estimate was made of the conditional probability of core melt given the specific damage state induced in the Browns Ferry systems by the fire (USNRC, 1975). The unconditional frequency of fire-induced core melt, calculated by averaging out the observed frequency of the Browns Ferry type of fire over the experience of U.S. commercial nuclear power plants, was found to be 1×10^{-5} per reactor-year, which is about 20 percent of the total core-melt probability estimated in the Reactor Safety Study. Kazarians and Apostolakis (1978) performed the same type of calculations under different assumptions and concluded that the frequency of core melt could be higher by a factor of 10. Both of these analyses appropriately point out that the results apply only to the specific circumstances of one particular fire and should not be construed as an estimate of the total contribution of fires to risk.

A more detailed risk analysis of fires was included in the Clinch River Breeder Reactor (CRBR) Risk Assessment Study (1977). A failure modes and effects analysis was used to identify important fire locations for a wide variety of combustibles, including cables, oil, and sodium. Its estimate of the frequency of fire-induced core melt, 5×10^{-7} per reactor-year, is substantially below the estimates discussed above. At least part of the difference in the estimates can be attributed to the vastly greater physical separation of cables and equipment in the CRBR design.

Further contributions to the risk analysis of fires were made in a risk-assessment study for the high-temperature gas-cooled reactor (HTGR) (Fleming et al., 1979). In addition to a qualitative screening procedure similar to that employed in the CRBR study, the HTGR study made use of a quantitative bounding method to screen for important fire locations.

Nuclear plant experience data were analyzed in detail to obtain estimates of fire-occurrence frequency and probability distributions for fire severity as measured by the duration of burn and the size of the fire-damage area. These data were used in a simple fire-propagation model to estimate the probabilities of location-dependent common-cause failures. A major finding of this study, which is independent of the unique characteristics of HTGRs, is that the contribution of fires to risk cannot be expressed simply in terms of core-melt frequency, as in earlier studies, because the conditional probability of containment failure given a core melt may be greater for fires than for other initiators. Fire-induced core-heatup accident sequences were found to dominate the HTGR risk-assessment curve at accident frequencies below 1×10^{-7} per reactor-year.

The Rensselaer Polytechnic Institute examined, for the U.S. Nuclear Regulatory Commission, various aspects of fire risk for light-water reactors. In Gallucci's work (1980) nuclear plant data are analyzed and categorized in the HTGR study. The source of data extended beyond licensee event reports to include insurance company records, and therefore the sample size was somewhat larger. The result was a more complete data base, particularly with regard to fires during construction.

In his doctoral dissertation at Rensselaer, Gallucci (1980) developed a risk-analysis method and applied it to a representative design for a large BWR. The probabilistic aspects of fire propagation were modeled in terms of an event tree that explicitly models various stages of ignition, detection, suppression, and propagation. The application of this technique is described in Section 11.3.3. The frequency of core damage due to fires was estimated to be about 2×10^{-4} per reactor-year, with an upper bound of about 1×10^{-3} per reactor-year. In this study, three types of combustibles at each of 11 plant locations were analyzed in the quantification of risk.

Recent advancements in the risk analysis of major fires have been made by Apostolakis, Kazarians, and Siu in projects carried out at the University of California at Los Angeles and as part of the Zion (Commonwealth Edison Company, 1981) and Indian Point (PASNY, 1982) risk studies.* Specific advancements in this work include the development of a physical model for fire propagation and suppression, a method for propagating uncertainties through this model, and the use of Bayes' theorem in estimating plant-specific and location-specific fire-occurrence frequencies. The Zion, Indian Point, and Big Rock Point studies have included detailed analyses of fire-induced accident sequences.

The trend in the risk analysis of fires is clear. There is a growing body of evidence to suggest that fires cannot and should not be dismissed as important risk contributors on a generic basis. In certain applications of risk analysis, such as those performed during the conceptual or detailed design stage, it may not be practical to attempt a fire-risk analysis

*See Apostolakis and Kazarians (1980), Apostolakis et al. (1982), Kazarians and Apostolakis (1978, 1981), Siu (1980), and Siu and Apostolakis (1981).

because of the need for factoring in details of the physical layout and construction. However, whenever the results of a risk study are to be interpreted on an absolute scale, the omission of fires appears to create a high risk of overlooking potentially dominant accident sequences. Fortunately, the methods discussed in this section include those that allow most fires to be screened out without the need for detailed investigation.

11.3.2 OVERVIEW

The purpose of the analytical method developed in the next section is to identify a list of the dominant accident sequences that are initiated by fire and then to assess the frequency of occurrence for each. This process requires information about several important aspects of a fire (e.g., ignition, progression, detection and suppression, characteristics of materials under fire conditions) as well as the plant safety functions and their behavior under accident conditions. Considerable uncertainties exist in the analysis because of gaps in the required knowledge. Nevertheless, the current analytical technique shows significant improvements over that available only a few years ago.

Fires are generally treated as external events, although they are generated by plant equipment and personnel. It is assumed that fires are initiated at certain frequencies within various plant compartments; the analyst is to determine what sequences follow and with what frequency.

Following the standard format for the analysis of external events, the fire analysis is divided into four parts: a hazard analysis; a fire-propagation analysis, which is somewhat analogous to a component-fragility analysis; a plant and system analysis; and a release-frequency analysis. The hazard analysis develops the frequency and magnitude of the "externally imposed stress," where "stress" is in terms of potential fire-induced accident sequences. The propagation analysis investigates the resistance of the plant to fire damage by studying the propagation of the fire and the effectiveness and timing of suppression. The last two analyses evaluate the response of plant systems to the accident sequence triggered by the fire; the first considers core damage, while the second is concerned with the release of radioactive material from the containment. This division is by no means unique; it simply provides a familiar structure that can be used to examine the methods of fire-risk analysis, which are described below.

11.3.3 METHODS

The object of the fire-risk analysis is to estimate the frequency of fire-induced radionuclide releases of varying magnitudes. Because of the inherent variability of fire phenomena and the relatively primitive understanding of these phenomena, the large uncertainties in the models leading to these release-category frequency estimates should be treated explicitly throughout the analysis.

The method can be divided into four somewhat independent steps: the fire-hazard analysis, which identifies critical plant areas and estimates the frequency of fires; the fire-propagation analysis, which models the behavior of fires in the critical areas; the plant-system analysis, which estimates the likelihood of the fires leading to plant-damage states; and the release-frequency analysis, which uses the results of the preceding analyses to derive the frequencies of accident sequences leading to radionuclide releases.

The sections that follow discuss the merits of various models available for each analysis.

11.3.3.1 Fire-Hazard Analysis

11.3.3.1.1 Location Screening

Theoretically, the fire-risk analyst should study the potential contributions to risk of fires anywhere in the nuclear power plant. By screening out unimportant locations, however, he can greatly reduce the amount of work required without sacrificing significant confidence in his results. The purpose of the fire-hazard analysis is to identify the locations that are important to the fire-risk analysis.

For the purposes of initial analysis, fire locations are usually considered to be coincident with the fire zones defined by the utility in its fire-protection review, issued in response to the NRC's Technical Position 9.5-1. The fire zones consist of one or more compartments and are separated from other zones by rated fire barriers. The spread of fire between zones is generally unlikely, although flames did spread through an improperly sealed cable penetration in the well-known Browns Ferry fire. A more detailed analysis may show that only limited areas within the fire zones contain critical equipment (Commonwealth Edison Company, 1981), in which case a number of fire locations may be defined within these zones.

The "importance" of a fire location is measured by its contribution to the frequency and the nature of a release of radioactive material. Since this cannot be determined until at least the first iteration of the fire-risk analysis has been completed, more approximate measures are employed. The primary measures are the type and the quantity of fire-vulnerable safety equipment at the location of interest. This information can be obtained directly from the fire-protection reviews. Other factors that may be used in the screening process are the frequencies of fires, the types and the amounts of combustible materials, and the available fire-suppression systems. Information on the last three factors can also be obtained from the fire-protection reviews.

Three basic methods for determining the importance of plant locations with respect to fire risk are described below. The first considers the presence of fire-vulnerable safety equipment, the second employs a failure modes and effects analysis (FMEA), and the third uses an FMEA coupled with additional factors to account for the likelihood of severe fires.

Location Analysis: Method 1

The location of interest is considered important if it contains enough safety equipment so that a severe fire could fail one or more safety systems (e.g., shutdown heat removal), which may or may not be in the same division. The loss of only one division of safety equipment means a loss of redundancy and does not necessarily lead to core damage and a release of radionuclides; nevertheless, the analyst may decide that this is an event that should be quantified.

The fire-protection reviews and a recent study at the Rensselaer Polytechnic Institute (Gallucci, 1980) employ such a screening approach. Because there are many rooms that contain some safety equipment, these studies consider fire occurrences in many locations. However, only a small number of these locations contribute significantly to the risk in most power plants: the rooms that contain many divisions of safety equipment. The less critical locations are screened out by performing an FMEA.

Location Analysis: Method 2

As in method 1, the locations containing fire-vulnerable safety equipment are identified. The loss of all equipment at that location is then postulated. If it is found that an initiating event (LOCA or transient) will not occur, the location is eliminated from consideration. (Note that a reactor trip, which is a transient by definition, will almost certainly be induced by a fire severe enough to disable many items of safety equipment.) Given a LOCA or a transient, a number of safety functions are required for safe shutdown. If the loss of all equipment in the location of interest prohibits the performance of any or all required functions, the location is tabbed for further analysis.

This screening method, described by Kazarians and Apostolakis (1981), was employed in the fire-risk portions of the Zion (Commonwealth Edison Company, 1981) and Indian Point studies (PASNY, 1982). In these analyses, the fire-induced loss of control of safety systems is judged to dominate fire-induced hardware losses, and therefore the critical locations that were investigated contain electrical cables and/or switchgear for many safety (and nonsafety) systems.

Given this assumption, and a fire that has caused an initiating event, the analysts then determine whether the same fire can induce failures that will prevent--

1. Reaching and maintaining a condition of negative reactivity.

2. Removing decay heat.

3. Monitoring and controlling the inventory and pressure of the reactor-coolant system (RCS).

Because of the fail-safe logic of the reactor-scram circuitry, the analysts assume that the reactor trip is successful. Therefore, critical fires will affect the implementation of actions 2 and 3. If no one fire can do this, the fires that can prevent either action 2 or action 3 are considered. However, these fires are less likely to be significant contributors to risk, because at least one independent failure in an unaffected safety system is required for core damage and radionuclide release to occur.

One feature of method 2, as executed by Kazarians and Apostolakis (1981), is that sets of fire locations are not considered for evaluation. For instance, if two adjacent rooms each contains one train of safety equipment, fires that spread from one room to the other and disable both trains are not studied. It is felt that a very large and long-burning fire is needed to penetrate most power-plant compartment walls, whether or not they are fire-rated walls. These fires are low-frequency events, and their contributions to risk are likely to be dominated by the contributions from fires burning in rooms that contain both trains.

If interzone fire propagation is considered to be important, a more complicated screening procedure that looks at groups of adjacent locations can be employed. One such approach uses a component-level fault tree in which the components are assigned location identifiers. "Core melt" is typically the top event, although it need not be. Minimal cut sets for the fault tree are derived in terms of the location identifiers. Several computer codes are available for this purpose. (For a discussion of such qualitative search procedures, see Section 3.7.) The WAMCOM code (Putney, 1981) is especially useful because it can identify cut sets with up to two locations.

One disadvantage of this method is the potential for omitting fire sequences leading partway to core meltdown but requiring additional component failures to result in the top event. For example, a sequence initiated by a fire in one location plus a dependent failure of components in each of two other locations would be screened out by this method, although it is questionable that the probability of such a sequence is really very low.

The screening approach of method 2 is illustrated here by example. The location of interest is the outer cable-spreading room of an imaginary power plant. The analyst must determine whether a fire in this location can cause not only an initiating event but also prevent the removal of decay heat, prevent the monitoring and control of RCS coolant inventory and pressure, or do both.

The outer cable-spreading room contains control and power cables for the motor-driven pumps of the auxiliary feedwater system, for the power-operated relief valves, and for the safety-injection pumps; control cables for the containment sprays and fan coolers; and control and power cables for many other systems.

A fire in this room has the potential to cause a LOCA by spuriously activating an isolation valve. If this does not happen, the presence of a large number of control and instrument cables in the room virtually ensures that a transient will occur. Thus, the first screening criterion of method 2 is met. Since the loss of the equipment mentioned above will clearly affect the removal of decay heat and the control of the RCS coolant inventory and pressure, criteria 2 and 3 are satisfied at least partially, and so this room is a candidate for an in-depth analysis. The presence of control cables for the containment sprays and fan coolers further emphasizes the room's importance: a fire that leads to core damage can also inhibit the performance of these containment-protection systems.

Of course, the cable-spreading room is an intuitive choice to begin with, not to mention the fact that the Browns Ferry fire involved this room. Most of the locations selected by this screening method are indeed "obvious" danger spots. However, the method provides a systematic means for their selection, sometimes identifies rooms that are not quite so obvious, and even rejects rooms that may seem to be obvious choices but actually do not contain enough critical equipment.

A more serious criticism is that this method does not account for the possibility that the fire-caused simultaneous failures of many instruments and/or nonsafety systems may initiate accident sequences. One might question the importance of this weakness when a power plant has single rooms that contain redundant safety trains and is vulnerable in the manner considered above. Furthermore, this weakness is common to all screening procedures described here and to the overall methods that are currently used for fire-risk analyses.

<u>Location Analysis: Method 3</u>

This method employs additional measures of importance to supplement either method 1 or method 2. In one model of this type, each room's fuel loading, fuel type, and fire-suppression effectiveness are combined with the safety-equipment inventory by a judgmental ranking procedure (Hockenbury and Yater, 1980).

A more elaborate fire location and progression analysis (FLPA) is described by Fleming et al. (1979). In addition to the effects of a fire on the components inside a room and the subsequent plant response, this method takes into account the inventories of combustible materials, the characteristics of adjacent locations, fire-brigade access, and ventilation systems; it also uses qualitative assessments of the likelihood of fire initiation and progression. Since the characteristics of adjacent compartments are explicitly considered, the possibility of fire spread from rooms containing large inventories of combustibles to compartments containing safety equipment is not overlooked.

Fleming et al. (1979) also discuss a method where the frequency of a particular radionuclide-release category due to all initiating events except fire, divided by the conditional frequency of that release category, given the loss of all components in the zone of interest, is compared with a rough estimate of the frequency of fires for that zone. If the release-category frequency ratio is greater than the fire frequency, the location is judged to be an insignificant contributor. This method requires a prior or concurrent assessment of other initiating events.

Clearly, the screening procedures of method 3 require more information and analysis than those of method 2. It is worthwhile to consider the merits of the additional complexity. The selection of a method and its implementation should be based on the objective of minimizing the chances that important fire-source locations will be overlooked in balance with the objective of minimizing the expenditure of effort on unimportant locations.

11.3.3.1.2 Fire-Occurrence Frequency

Once the analyst has identified the locations where a fire has the potential to initiate an accident sequence leading to a release of radionuclides, it is natural to ask how often these fires occur. Although an internal event, the fire is viewed as an external stress imposed on the plant at random times. The rate of occurrence can be established from the historical record.

Data from more than 900 reactor-years of U.S. experience are available for evaluation when construction and preoperational testing periods are included with plant operating periods. Considerable effort has been spent in evaluating the fire events cataloged in licensee event reports and data from the American Nuclear Insurers (see, for example, Hockenbury and Yater, 1980; Fleming et al., 1979). Data extracted from both sources are shown in Tables 11-2 and 11-3. The nature and the frequency of fires at nuclear power plants change dramatically, however, between construction, preoperational testing, and plant operation. Consequently, only the plant operating histories are suitable for assessing the risk from plants at power.

A number of issues arise in using the available data in estimating the rates of location-dependent fire occurrence. These include the possible reduction in the frequency of fires that results from an increased awareness

Table 11-2. Frequency of fires by reactor type

Reactor type	Status or mode of operation	Number of events	Percentage of total[a]
Boiling water	Construction	37	15.7
	Preoperational testing	6	2.5
	Operation	25	10.6
	Hot shutdown	0	0.0
	Cold shutdown	1	0.4
	Refueling/extended outage	3	1.2
Pressurized water	Construction	61	25.9
	Preoperational testing	15	6.4
	Operation	25	10.6
	Hot shutdown	4	1.7
	Cold shutdown	4	1.7
	Refueling/extended outage	1	0.4

[a]Total includes 22 events in fuel-fabrication facilities, 1 event in a reprocessing plant, 27 events in research and educational reactors, 2 events in a high-temperature gas-cooled reactor, and 3 events in a fast breeder reactor.

Table 11-3. Summary of fire-experience data

Number of reactor units	65
Operating experience (reactor-years)[a]	372
Number of fires (in operation)	49
Mean rate of occurrence per reactor-year	0.13
Diameter of fire damage (ft)	
Mean	8.6
Maximum	67
Time to put out fire (hr)	
Mean	1
Maximum	24

[a]From first electricity generation through April 1978.

of the danger of fire (Apostolakis and Kazarians, 1980; Gallucci, 1980), the discrepancy between the actual number of fire occurrences and the number of reported fires (Hockenbury and Yater, 1980; Hockenbury et al., 1981), and the question of applying industry-wide fire data to a particular power plant (Commonwealth Edison Company, 1981). These problems give rise to large uncertainties in the interpretation of the historical data.

Apostolakis and Kazarians (1980) model the frequency of fires for various compartments, using a probability-of-frequency framework to consistently treat the uncertainties. Starting with broad prior distributions to model their weak state of knowledge, they employ the statistical evidence given in Table 11-4 and use Bayes' theorem to derive the fire frequencies shown in Table 11-5. Their procedure is roughly outlined below.

The gamma probability distribution, with the parameters α and β given in Table 11-5, is chosen to represent the prior distributions for the various compartments. The gamma distribution is

$$\Pi(\lambda) = \frac{\beta\alpha}{\Gamma(\alpha)} \lambda^{\alpha-1} \exp(-\beta\lambda)$$

where $\Pi(\lambda)$ is the probability density function of fire frequency λ. The likelihood of the data, the probability of γ fires in T reactor-years (see Table 11-4), given fire frequency λ, is modeled as

$$L\left(\frac{E}{\lambda}\right) = \exp(-\lambda T) \frac{(\lambda T)r}{r!}$$

Bayes' theorem then states that the updated probability distribution for fire frequency, given evidence E, is

$$\Pi'\left(\frac{\lambda}{E}\right) = \frac{\Pi(\lambda)\ L(E/\lambda)}{\int_0^\infty d\lambda\ \Pi(\alpha)\ L(E/\lambda)}$$

Table 11-4. Statistical evidence of fires in light-water reactors[a,b]

Area	Number of fires, r	Number of relevant years, T
Control room	1	288.5
Cable-spreading room	2	301.3
Diesel-generator room	10	543.0
Containment	5	337
Turbine building	9	295.3
Auxiliary building	10	303.3

[a]From Apostolakis (1980).
[b]As of May 1, 1978.

An analytical evaluation of this expression shows that $\Pi'(\lambda/E)$ is also a gamma distribution, but with characteristic parameters $\alpha' = \alpha + r$ and $\beta' = \beta + T$. These updated values are also given in Table 11-5. For example, in the cable-spreading room from Table 11-5, the values of α and β (0.182 and 0.96) yield a mean frequency of .21, while the posterior distribution α' and β' (2.182) and 302.26) yields a mean frequency of .0072.

This same procedure can be used to update the given distributions for fire frequencies, when further reactor experience is accumulated.

Table 11-5. Distribution of the frequency of fires[a]

Area	Parameters of gamma distribution α	β	Frequency of fires per room-year λ_{05}	λ_{50}	λ_{95}	$\langle \lambda \rangle$
Control room						
Prior	0.182	0.96	5.0×10^{-8}	0.015	1.0	0.21
Posterior	1.182	289.46	3.1×10^{-4}	0.003	0.012	0.0041
Cable-spreading room						
Prior	0.182	0.96	5.0×10^{-8}	0.015	1.0	0.21
Posterior	2.182	302.26	1.4×10^{-3}	0.0062	0.017	0.0072
Diesel-generator room						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	10.32	543.29	1.1×10^{-2}	0.018	0.03	0.019
Containment						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	5.32	337.29	6.2×10^{-3}	0.014	0.028	0.016
Turbine building						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	9.32	295.59	1.7×10^{-2}	0.03	0.05	0.032
Auxiliary building						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	10.32	303.59	1.9×10^{-2}	0.033	0.053	0.034

[a]From Apostolakis (1980).

It should be noted that the fire-frequency distributions derived are generic. To account for plant-to-plant variability, similar distributions based on each plant's experience can be constructed, either by using the above procedure or by updating the generic distributions in Table 11-5.

11.3.3.1.3 Fire-Propagation Analysis

The purpose of a fire-propagation analysis is to determine the likelihood and extent of various levels of damage in a compartment, given that a fire has occurred. Three different approaches have been used to date. The first employs a statistical model based on past experience (Fleming et al., 1979), the second uses a multistage event-tree model (Gallucci, 1980), and the third requires the construction of physical models (Siu, 1980; Siu and Apostolakis, 1981).

The analyst should be aware that the existing fire-growth and fire-suppression models do not span the set of all possible scenarios and that even the existing models exhibit large uncertainties. Every attempt should be made to quantify the effects of these uncertainties.

Fire-Propagation Analysis: Method 1

This method is based on deriving equations for (1) the distance of fire spread or volume affected versus the time to fire control and (2) the probability of control versus the time to fire control (Fleming et al., 1979). From the two equations a curve for conditional probability versus fire size can be obtained. Here "fire size" is defined as that size within which components are failed. The equations are derived from linear regression analyses of fire data from nuclear power plants. Different correlations are developed for different combustibles (e.g., electrical fires versus lubricating-oil fires).

Method 1 is relatively easy to implement and with some conservative assumptions can be very effective at screening out unimportant locations. It has not yet been applied to fires that penetrate fire barriers. This approach glosses over the specifics of plant design: it assumes that average fire-occurrence frequencies derived from the operating histories of many plants apply to the plant under study. In its application so far, method 1 has assumed that fire has an equal probability of starting anywhere in the location studied--as would happen if transient combustibles were the dominant sources of most fires or if the permanent combustibles were uniformly distributed throughout the location.

Fire-Propagation Analysis: Method 2

Method 2 uses event trees (Gallucci and Hockenbury, 1981) to divide the fire model into four elements: (1) ignition, (2) detection, (3) suppression, and (4) propagation. Each element heads a column of the event tree. The fire is assumed to start in one component and potentially propagate to the next. The use of these four elements is illustrated in Figure 11-11 (Gallucci, 1980) by a two-stage event tree for two redundant components in the location (more stages may be required for more components). Submodels,

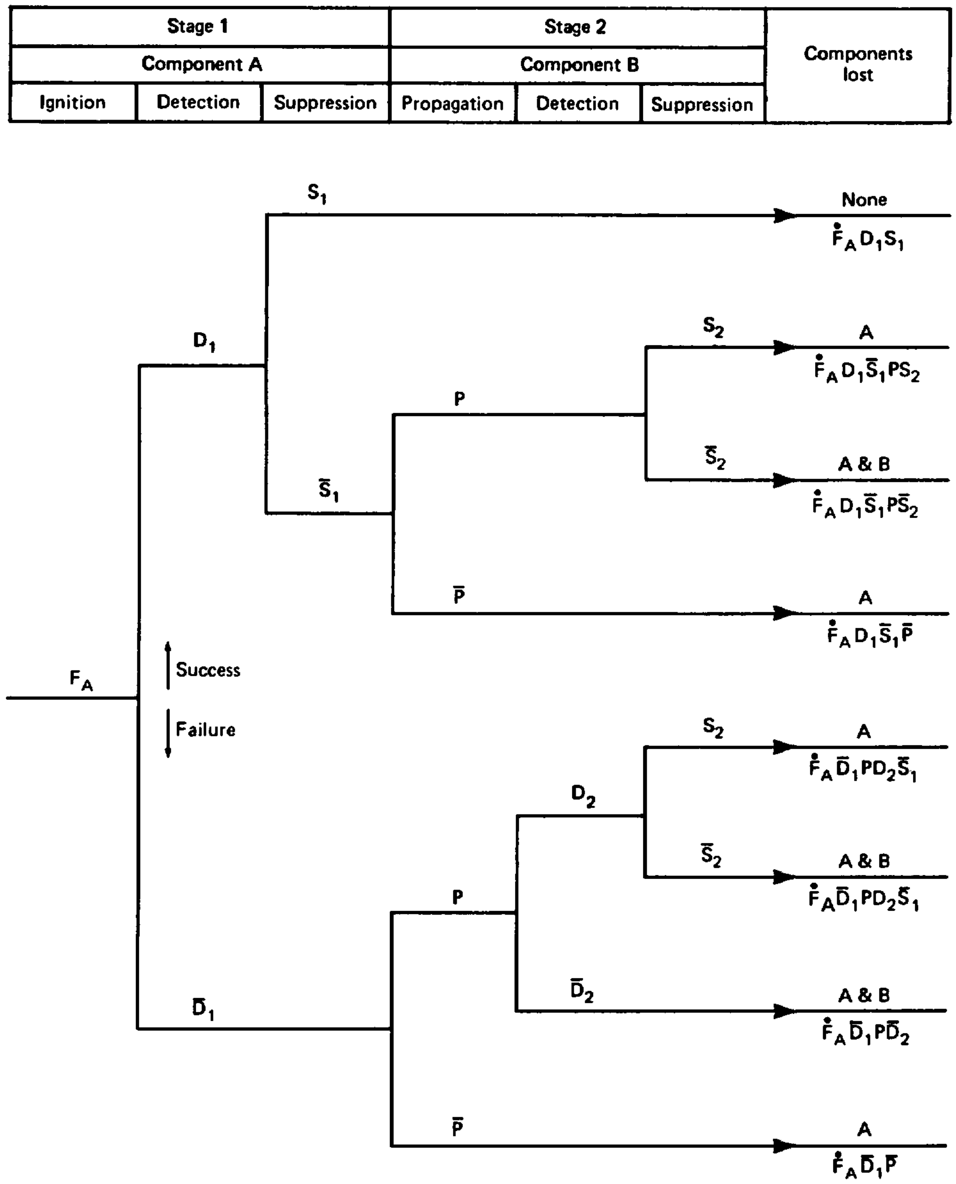


Figure 11-11. Illustrative two-stage event tree for two redundant components.

sometimes employing fault trees, are used to quantify the conditional branching probabilities of the event tree. Plant-design specifics, particularly for the detection and suppression elements, are accounted for in the submodels, as warranted. Much effort is placed on establishing the reliability of the fire-protection system (Moelling, 1979). Automatic and manual means of detection are included in the detection submodel.

Fire-Propagation Analysis: Method 3

In this approach, fire growth and suppression are viewed as competing time-dependent processes (Siu, 1980; Siu and Apostolakis, 1981). One or more representative fire-growth scenarios are developed for each location, depending on the physical configuration of the area. The distribution for the analyst-defined characteristic spread time is then compared against the distribution for the suppression time, to obtain the conditional frequency of fire growth, given the fire scenario.

For example, assume that two horizontal cable trays, one stacked over the other, contain critical power and control cables. In the representative fire scenario, a fire initiated on the lower tray spreads to the upper one in t_g minutes. The mean fire-suppression time is t_s minutes. Note that t_s includes the time to detect the fire, which often requires human response. Therefore, the distribution of the fire-spread frequency is the distribution of the frequency with which t_s exceeds t_g. The fire-spread time is computed by using physical models, while t_s is estimated from statistical data; their distributions describe the state of knowledge concerning fire processes.

The keys to this approach are the explicit use of simple physical models for fire (Siu, 1980), which enables the analyst to properly account for the extremely strong dependence of fire behavior on the physical configuration of the fuel bed and its surroundings, and the consistent treatment of the large uncertainties in the model outputs.

The simple physical model (Siu, 1980) is used to calculate the heat transferred from a fire to its surroundings, the time to ignition or damage for affected materials, and the subsequent rate of fire growth. Its predictions are subject to uncertainty, of course, because of statistical uncertainties in the behavior of fires, uncertainties caused by basic modeling assumptions, and uncertainties in the numerical values of the input parameters. The last-named source of uncertainty is propagated through the model by response-surface techniques, and the statistical uncertainties are often left unquantified, since they are generally dominated by the state-of-knowledge uncertainties. To treat the basic modeling uncertainty, the output of the model is treated as an expert's opinion, and a probability distribution for the accuracy of the model is constructed, based on available data and the judgment of the analyst.

The physical model of Siu (1980), called the "deterministic reference model," or DRM, was used in the Zion (Commonwealth Edison Company, 1981) and Indian Point (PASNY, 1982) studies. It focuses on predicting radiative and convective heat transfer from a fire to an object. This object may be another portion of the fuel bed, a noncombustible component, or even a fire barrier. In the last case, the heat flux leaving the barrier is also computed, to determine the effect on objects that are not directly exposed to the flames. Using a simple ignition- or damage-threshold temperature criterion, the impact of the fire on its surroundings is then computed as a function of time.

For many configurations, the DRM consists of only a few equations, and the needed calculations can be done by hand. For more complex configurations, where the interaction of several burning fuel elements is important, the computer code COMPBRN (Siu, 1980) is useful.

COMPBRN was developed to analyze fairly general fire-growth scenarios in a compartment. Widely varying configurations, fuel types, initiating-fire characteristics, and room properties can be studied. Essentially, the primary limitation on the complexity of the simulation is economic, since the storage requirements of COMPBRN increase greatly with the number of fuel elements modeled. COMPBRN has been used to model a number of small experimental fires, with generally good results (Siu, 1980). Its predictions for very large fires approaching flashover may be subject to greater uncertainties.

The distribution for the fire-suppression time is estimated in the Zion and Indian Point studies from information presented by Fleming et al. (1979). This distribution is somewhat dependent on the size of the fire, the degree of dependence being assessed judgmentally.

Because the behavior and the effects of fire do depend strongly on the layout of the location of interest, the physical modeling of method 3 appears to be the favored approach for modeling fire propagation. Uncertainties in the estimates obtained with models like COMPBRN will decrease when the models are upgraded to reflect experimental results and advances in fire research field.

11.3.3.2 Plant-System Analysis

Once the frequencies of fire-induced component losses are assessed, it is possible to estimate the frequency of fire-initiated accident sequences leading to core damage.

As with other initiating events, separate event trees may be constructed for fires because the operator, rather than automatic actions, may be responsible for shutting down the plant in response to a fire. Often the analyst simply modifies the front end of existing event trees for other initiating events to specialize them for fires. The postulated fire can be assumed to coincide with another initiating event, in which case the original event-tree structure would be retained. The conditional branching probabilities would be altered to reflect the dependence on the fire. However, if fires are to be treated as a separate event, care should be taken that data from which basic component-failure rates are determined do not double-count these failures from fires.

As noted above, human intervention plays an important role in the accident. Not only can the operators extinguish the fire and operate equipment manually, they may make repairs and jury-rig replacement equipment as well. Then again, they may be misled by fire-caused faulty information and may actually exacerbate the situation. A variety of operator actions are modeled in the Zion and Indian Point fire-risk analyses, depending on the scenario considered, but the modeling is extremely crude at this stage. Some other issues that have to be addressed in the analysis of fire-induced accident sequences are smoke propagation, effects of fire-suppression activities, fires outside the plant, and the failure of fire barriers.

The quantification of accident sequences involving fires follows the lines described in the more general discussion of accident-sequence quantification; the interfaces are described in Section 10.3.6. Particular attention must be paid to the intersystem dependences introduced by fires. Fires as causes of component failures may be included as "house" events directly in the system fault trees as a function of location and size. System reliabilities can then be evaluated conditional on the occurrence of the postulated fire.

Another point to be emphasized is that the uncertainties in the analysis of fire frequency and propagation must be combined with the event-tree uncertainties. A rational comparison of various sources of risk (e.g., fires, floods, hardware, etc.) requires the recognition and consistent treatment of uncertainties.

Once the sequences involving fires are delineated and frequency distributions quantified, the assessment of plant-system response proceeds as with other initiating events. Besides direct impacts on system components, fires have other deleterious impacts: the flooding that results from attempts to extinguish the fire; smoke, which may hinder personnel access; the generation of ignition sources for other flammable products; the possible boiling of water inside pipes passing through the fire; and the like. These impacts have not been addressed in detail by any of the fire studies to date. The dependences of fire as a secondary event to some other external event (e.g., fire initiated by an earthquake) have also not been evaluated, although the methods described here and in Chapter 10 are fully applicable to these dependences.

11.3.3.3 Release-Frequency Analysis

The purpose of this analysis is to derive the distributions for the various categories of radionuclide releases from the containment. The comments of the preceding section apply here as well, although the end result of the accident sequences is release rather than core damage. The release-category analyses should take into account that the same fire that damaged the reactor core may well have damaged containment mitigating functions also. A careful investigation of the entire accident sequence, and not just the portion following core damage, is required.

11.3.4 INFORMATION REQUIREMENTS

The information required to perform a risk analysis of fires can be summarized as follows:

1. Description of plant systems, including the location of components and systems within structures. Especially important are routings for safety-related power and control cables.

2. Fire-protection report, which contains information on transient and permanent fuel loadings, suppression systems, ventilation systems, and safety-equipment inventories for each fire zone as well as a simplified FMEA for some zones.

3. Reports on the fire qualification of components. Physical data for electrical cables and trays are very useful.

4. Results of the plant-system analysis for internal initiating events, especially accident-sequence descriptions. Accident-sequence frequencies are also useful for screening purposes.

5. A compilation of licensee event reports involving fires at nuclear power plants.

11.3.5 PROCEDURE

The methods described in Section 11.3.3 are summarized below in the form of a task-by-task procedure for performing a risk analysis of fires.

Task 1: Fire-Hazard Analysis

1. Construct simple systems model of plant.

2. Identify locations of safety equipment.

3. Identify critical fire-impact locations, using a simple FMEA.

4. Identify locations adjacent to critical locations containing large quantities of combustibles.

5. Evaluate the distributions for fire frequency for each location.

Task 2: Fire-Propagation Analysis

1. Define representative fire-growth scenarios for each location.
2. Determine distribution for fire-growth time for each scenario.
3. Determine distribution for fire-suppression time for each scenario.
4. Compute distribution of frequency of growth.

Task 3: Plant and Systems Analysis

1. Develop event- or fault-tree logic that links component damage to one or more core-damage states.

2. Apply component and system failure boundary conditions to the event- or fault-tree logic.

3. Develop the distributions for the frequency of fires resulting in each core-damage state.

Task 4: Release-Frequency Analysis. Proceed as in Task 3, but carry out to release categories.

Task 5: Iterate.

11.4 RISK ANALYSIS OF FLOODS

11.4.1 INTRODUCTION

This section describes methods and procedures for assessing the consequences of reactor accidents involving external or internal floods. The methods and procedures described here should be used in conjunction with the external hazards screening criteria described in Chapter 10 and are conceptually similar to those presented in Sections 11.2 and 11.3 for the analysis of seismic and fire risk, respectively.

In comparison with some other external hazards, particularly fires and earthquakes, floods have received less attention as a potential cause of reactor accidents in the PRA studies carried out so far. As a consequence, there are no well-established methods for the analysis of either external or internal floods. The implied perception is that floods are less likely than fires and earthquakes to induce accidents that might contribute significantly to the overall risk of a nuclear plant. This perception is supported by the view that the flood-protection measures required for licensing have resulted in extremely low frequencies of floods that produce significant damage. In addition--and this is especially true of most external floods--even when floods of extreme severity are postulated to occur, there should often be ample warning time to safely shut down the reactor before significant damage in important systems and structures can occur.

However, there are several reasons for not excluding floods as potentially important risk contributors in PRA studies. First, there are large uncertainties in the estimated frequencies of external floods of extreme severity and in the associated fragilities of plant structures and components. Second, some causes of flooding, such as the failure of an upstream dam or a large rupture, inside the turbine building, in the circulating-water system may not provide significant warning time. Third, many of the design and operational features required to protect against external floods may not provide the same degree of protection against internally initiated floods. In fact, the experience with flooding at nuclear power plants indicates that internal floods may have a relatively greater potential to cause a reactor accident with nonnegligible risk. For example, Table 11-6, which is taken from a report by Verna (1982), summarizes the U.S. nuclear experience with turbine-building floods from internal sources. One of the more serious of these internal flood events is described in the paragraph that follows.

In June 1972, at Quad Cities Unit 1, a rupture in the circulating-water system caused the rapid flooding of a room containing a number of pumps in different systems. The equipment damaged by the flood included four service-water pumps for residual heat removal, two diesel-generator cooling-water pumps, four condensate-booster pumps, and three condensate-transfer pumps. In addition, the floor-drain sump pumps, the hypochlorite system analyzer, and condensate-pressure gauges were damaged. Although the reactor was not damaged, the impact of this flood in terms of the failure of multiple components and systems was extensive. Modifications were made at Quad Cities to enhance the physical separation of the safety-related pumps and thus protect against the recurrence of a flood in the same room.

Table 11-6. Turbine-building flooding in U.S. nuclear power plants[a,b]

Date of occurrence	Plant	Affected safety component	Spill rate	Remarks
		SOURCE: SERVICE WATER		
June 1975	Surry 2	Service-water valve		Pump developed seal leak
October 1977	Surry 2	Service-water valves of all redundant trains		Personnel forgot to close valves that were opened for maintenance
October 1978	E. Hatch 1	Service-water valve		Valve body blew out during repair
October 1979	Dresden 2	Diesel-generator control cabinet		Fire-water leak
		SOURCE: CONDENSER CIRCULATING WATER		
January 1979	Crystal River 3		Large	Solenoid valve failed open and led to flooding
April 1977	Three Mile Island		Large	Circulating-water pump casing split 360°
October 1976	Oconee	Emergency feed-water pumps	Large	Pneumatic isolation valve opened when condenser man-hole was open and spilled lake water into turbine building
October 1978	Surry 2	Service-water valves	Small	Intentionally flooded during maintenance
June 1972	Quad Cities 1	Many redundant and diverse safety-related components	Very large	Valve closed inadvertently, and water hammer ruptured expansion joint

[a]Data on incidents from the start of commercial power operation up to July 1981.
[b]From Verna (1982).

As a result of NRC followup, various modifications were also made at 10 other plants to enhance protection against the flood-induced loss of safety functions (Verna, 1981).

A similar flood occurred at Three Mile Island Unit 1 in April 1977; it was caused by a leak in the circulating-water system at the casing of one of the circulating-water pumps. However, because of the plant's layout, damage was confined essentially to the six circulating-water pumps and did not affect any other systems (Verna, 1981).

These events and other incidents involving flooding indicate that, at least for certain nuclear power plants, internal floods may be an important cause of multiple, dependent failures. It is also apparent that differences in design features, such as provisions for physical separation, and plant layout can give rise to significant differences in the plant's response to the same flooding condition.

In summary, there are a number of reasons why flooding from external and internal causes should be considered for analysis in PRA studies. The operating experience of reactors includes floods that have resulted in the coincident loss of multiple components and even multiple systems. Attempts to estimate the frequency of severe external floods have resulted in the identification of large uncertainties. Finally, flooding is one of the failure mechanisms associated with common-cause failures--that is, multiple concurrent failures due to the same cause--which have long been recognized as an important factor in risk assessment (see Section 3.7).

Because of the relatively low emphasis given to external events in general and to floods in particular, the guidance given here on assessing the risk associated with flooding does not benefit from the same degree of experience in the development and application of PRA methods as that set forth for transients and events that initiate LOCAs. Among the external events, floods rank behind earthquakes and fires in terms of the PRA-relevant work that has been carried out.

This is not to say, however, that the current state of the art is insufficient to quantify the risk associated with flooding. As will be shown in this section, the analysis of such risk can be structured around the same basic hazard-fragility-systems approach that has been successfully applied to earthquakes, missiles, and other external events. In addition, some of the methods used in location-dependent common-cause analyses of fires are applicable to floods as well. The existing methods for calculating the frequencies of flood-induced accident sequences (especially for external floods) have resulted in wide uncertainties. It is recommended that, regardless of the magnitude of the uncertainty, the PRA should include a flooding analysis and attempts should be made to quantify the effects of uncertainties to the extent that this can be done. For these reasons, this section of the procedures guide was set aside for the risk analysis of flooding.

11.4.2 OVERVIEW

The probabilistic analysis of reactor accidents involving flooding can be viewed as a problem in determining $f_k(\underset{\sim}{z})$, the unconditional frequency of exceeding damage level $\underset{\sim}{z}$ of consequence type k, resulting from potential reactor accidents initiated by floods.

It is convenient to expand the external event risk equation (Equation 10-1) to the following form for the risk analysis of floods:

$$f_k(\underset{\sim}{z}) = \int_{\mathscr{D}} \cdots \int \sum_{j=1}^{J} \sum_{\ell=1}^{L} f_{E,\ell}(y) f_{S,j|E,\ell} f_{k|S,j}(\underset{\sim}{z}) h(\underset{\sim}{x}) d\underset{\sim}{x} \qquad (11\text{-}34)$$

where $f_{E,\ell}(y)$ is the frequency of flood-damage state E_ℓ given response $\underset{\sim}{y}$ to flood level $\underset{\sim}{x}$, $f_{S,j|E,\ell}$ is the frequency of accident sequence S_j given flood-damage state E_ℓ, and the quantities $f_k(\underset{\sim}{z})$, $F_{k|S,j}(\underset{\sim}{z})$, and $h(\underset{\sim}{x})\, d\underset{\sim}{x}$ are defined as in Equation 10-1.

Note that the magnitude ($\underset{\sim}{x}$) and response ($\underset{\sim}{y}$) are vectors to accommodate the multivariate aspect of floods. For example, a single flood event, such as a hurricane, can produce multiple effects, such as winds, wind waves, and high water levels.

Certainly, all the above defined parameters should be expressed by a probability distribution, depicting the uncertainties in the estimation processes and underlying data. These distributions can be derived by the methods given in Chapter 12 on the treatment of uncertainties.

Depending on the range of possible damage resulting from the causes of flooding under consideration, the term $\underset{\sim}{z}$ in Equation 11-34 could be either a discrete level of plant damage (e.g., core melt), a measure of the magnitude of a release of radionuclides, or an estimate of the health effects expected in the population at risk.

Risk analysis for flooding is performed along lines similar to those followed for other external events like earthquakes and fires. The steps include a flooding-hazard analysis, a fragility and vulnerability evaluation, a plant and system analysis, and a release-frequency analysis. A flooding-hazard analysis consists of identifying the site-dependent causes (e.g., dam failures) and associated failure mechanisms (e.g., submersion) and estimating the flood-hazard intensities $h(\underset{\sim}{x})$ for each flooding variable. Uncertainties in estimating the hazard intensities can be expressed by providing a family of curves, each with a state-of-knowledge probability assigned in the same manner as is done in seismic analysis.

The task of estimating the frequency of various flood-damage states for each failure mechanism, $f_{E,\ell}(\underset{\sim}{y})$, is referred to as the "analysis of flooding fragility and vulnerability." It entails the definition of a suitable set of flood-damage states that may include, for some external floods, damage to plant structures. In this case, it is necessary to estimate the frequencies of structural failures and associated state-of-knowledge probabilities that describe the level of uncertainty. The method of arriving at these estimates is much the same as that used in the seismic risk analyses described in Section 11.2, except that different failure mechanisms should be addressed, as will be explained below. If components are submerged, the fragility might reduce to a simple step function with a transition in the frequency of failure from zero to unity as the flood-height parameter reaches or exceeds the elevation of the component.

The tasks of plant and system analysis and release-frequency analysis are performed to complete the risk assessment of flooding and to quantify the terms $F_{S,j|E,\ell}$ and $f_{k|S,j}(\underset{\sim}{z})$. This is where the connection is made between the elements of the risk modeling that are unique to flooding and those generic to all the other initiating events analyzed in a PRA study, such as transients and LOCAs. Stated another way, the above terms include the event- and fault-tree logic that relates the various states of flood damage to the public risk from accidental releases of radioactivity. The details of this interface are discussed in Section 10.3.6.

If the particular flood in question has only plant-hardware implications and does not influence the calculated offsite consequences, as would happen in the case of an internal flood confined to a room, it may be desirable to carry out the flood risk analysis only to some intermediate "pinch point." One convenient pinch point is the frequency of occurrence of core damage or melt. However, if the flood is seen to influence the transport of radioactive material or to hinder the evacuation of people, an evaluation of flood-specific environmental consequences may be more appropriate.

It should be emphasized that a comprehensive evaluation of the risk from flooding--one that includes a complete quantification of each term in Equation 11-34--has not yet been carried out in a PRA for a nuclear plant. However, a substantial amount of technical work has been done in quantifying various elements of Equation 11-34 from which to synthesize an overall method for quantification; this work is summarized in Section 11.4.3. It is clear that more developmental research, as well as attempts at application, is necessary to bring the level of flood-risk analysis to the current state of the art for the analysis of seismic and fire risks.

Nonetheless, it would be a mistake to forego the inclusion of floods in a plant-specific risk analysis just because of the formative state of the methods, including those used for the probabilistic quantification of uncertainties. A relatively undeveloped method for quantification simply gives rise to greater uncertainties. From the perspective of enhancing design and licensing decisionmaking through risk analysis, the decision to include or exclude a candidate risk contributor should be based solely on its perceived contribution to risk. As already argued in Section 11.4.1, there is insufficient evidence to dismiss flooding as a potential risk contributor on a generic basis.

11.4.3 METHODS

A flood-risk analysis follows the general procedure described in Chapter 10. It consists of a flooding-hazard analysis, a component-fragility evaluation, a plant and system analysis, and a release-frequency analysis. Certain details, particularly in the areas of hazard and fragility analysis, are different, depending on whether the flood results from external or internal causes. Before describing these differences, it is instructive to briefly review the relevant literature and to set out the criteria for an acceptable probabilistic analysis of flood-induced accidents.

11.4.3.1 Relevant Literature

The first comprehensive assessment of accident risks in U.S. commercial light-water reactors, the Reactor Safety Study (RSS), considered certain external events. Unfortunately, the assessment of floods, as can be inferred from the presentation of the results (USNRC, 1975), did not include a quantification of this risk contributor, nor did it include a quantification of any of the terms in Equation 11-34. The qualitative assessment did, however, provide some insights of interest here. It showed, for example, that the basic PRA methods employed in the Reactor Safety Study are applicable to floods as well as to other external events. More specifically, the topology of accident sequences displayed in the RSS event trees was found to be applicable to floods. It also indicated that the frequency (referred to in the Study as "probability") of a core melt induced by floods at a river site should be expected to be extremely low--because structures enclosing essential safety-related equipment are specifically designed to survive a hypothetical flood called the probable maximum flood (PMF). Because of the conservatism in the way in which the PMF is calculated, it is argued that the frequency of the PMF is very low, as is the conditional frequency of failure for the associated structures. The qualitative assessment obtained from these calculations indicates that floods represent negligible risks. A similar conclusion and rationale are presented for coastal sites subjected to impulsive generated waves (tsunamis) and wind waves or the high water levels, waves, winds, and erosion due to tropical storms (hurricanes) and extra-tropical storms.

Wall (1974) has reviewed methods for estimating the frequency (return period) of floods at a river site. The approach emphasized in his paper is the statistical analysis of river-discharge data by means of various curve-fitting techniques. Different types of distributions are fitted to the same 44 years of river-discharge data, including log-Pearson type III, lognormal, and one of the extreme-value distributions, which is fitted by using maximum-likelihood estimators. Wall noted that excessive extrapolations of these curves beyond the range of the data would be required for an estimate of the frequency of floods approaching the magnitude of the PMF established for the site in question. An evaluation was made of the consequences of the PMF in terms of the warning time, damage to offsite-power supplies, and the role of watertight barriers. Wall concluded that the risk of a serious reactor accident due to rising water levels is negligible and proposed that the design-basis flood be redefined as that having a frequency of exceedence of 5×10^{-4} over the next 50 years, or 1×10^{-6} per reactor-year.

In addition to the need for excessive extrapolation, a major problem with estimating river-flood frequencies from historical data is the possibility that the historical data may have been rendered inapplicable by natural or man-made alterations to the hydrologic characteristics of the drainage basin. This observation has recently led to the suggestion of an alternative approach of calculating flood levels as a function of sequences of natural events and other factors relevant to flood levels (D. W. Newton, unpublished work). The statistical analyses are then performed on the subordinate events--intensity and duration of precipitation, relative sequence of successive storms, snowpack, temperature variations, and the like. Examples of an application to a sequence of rainstorms have been given by Alexander (1963). A model for predicting the frequency of hurricanes was developed by Mogolesko (1978). One of the limitations of Newton's approach is that statistical independence among the flood variables, often assumed in these applications, cannot be ensured without an extensive statistical analysis. For example, if successive rainstorms tend to cluster together in time, this assumption will be violated. The problem of changes in site hydrology affecting the applicability of historical flood data was also addressed in the development of the NRC's FLOE code, in which the curve fits to the data are subjectively modified by expert opinion applied with a Bayesian updating procedure.

One cause of river flooding that needs to be treated separately is the failure of an upstream dam. The frequencies and risks of earthquake-induced dam failures in California have been estimated in a study performed at the University of California at Los Angeles (Okrent et al., 1974).

Wagner et al. (1980) developed a method for estimating the effects of a river flood on the availability of systems. A computer code called NOAH was developed for estimating the fragility of a system--that is, the conditional frequency of failure--given the submergence of equipment at various flooding heights. The method was applied to the auxiliary feedwater system of Surry Unit 1, one of the plants analyzed in the Reactor Safety Study. The frequency of the initiating flood was not assessed in this study, but rather treated parametrically. It was found that, if the frequency of the postulated flood is assumed to exceed 10^{-4} per year, significant increases result in the estimated frequency of some of the dominant accident sequences identified in the Reactor Safety Study. Although several aspects of the flood equation (Equation 11-34) were not included--such as flood-hazard analysis, flood failure mechanisms other than submergence, and the quantification of uncertainties--this study is the best risk analysis of external floods performed so far, particularly with regard to how the terms $f_{E,\ell}(y)$, $F_{S,j|E,\ell}$ and $f_{k|S,j}(\underline{z})$ should be estimated.

In summary, the literature on flood-risk analysis includes statistical analyses of phenomena that contribute to floods, some qualitative assessments that indicate a low degree of risk from floods, and for external floods a computer-aided method for analyzing the location-dependent aspects of system fragilities. Although the literature does not yet include a comprehensive risk analysis of floods, several such studies are under way, including those for Midland and Oconee. There has been sufficient progress in specific elements of the methods to set forth criteria for an acceptable assessment of flood-induced accidents. These are presented below.

11.4.3.2 Acceptable Methods

In view of the current state of the art of PRA in general and flood-risk analysis in particular, it is recommended that the methods for analyzing the risk of flood-induced accidents meet the following criteria:

1. The methods should provide reasonable assurance that all sources of flooding, both external and internal, that are applicable to the site have been considered. Internal causes include leaks and breaks in major water systems, the overfilling of tanks, sump-pump malfunctions, and the backing up of drains. External causes include river flooding, dam failure, excessive precipitation, hurricanes, tsunamis, seiches, wind waves, and surges. Special attention should be paid to flood-protection provisions, their failure probabilities, and possible methods for flood termination (important to internal floods).

2. The methods should ensure completeness in the coverage of all relevant mechanisms of failure for structures and components that could affect risk. The following mechanisms should be considered at a minimum: loss of structural integrity through collapse, sliding, overturning, ponding, excessive impact and hydrostatic loads; flooding and wetting of equipment from seepage through walls and

roof; flow through openings; sprays, thermal shocks, missile impacts; and the blockage of cooling-water intakes by trash.

3. The analysis of flood frequency and flood-induced damage to plant structures, components, and systems must be properly integrated with the definition of accident sequences in the event trees, taking due account of the dependences associated with the flood. These dependences include the beneficial effects of warning time and the detrimental effects of increases in the frequencies of multiple concurrent failures.

4. In estimating the fragilities of structures and components, the failure criteria should be based on realistic assumptions and should not be considered synonymous with design limits.

5. In estimating the offsite consequences of flood-induced accidents, the impact of flood conditions on radionuclide transport and on evacuation should be considered. Regional emergency influences on the plant should also be taken into account, such as the loss of offsite power, the loss of communications, the loss of access, and various human factors. This is especially important for major floods from external causes.

6. The methods should ensure that all sources of uncertainty in the risk estimates are identified and their effects quantified if possible, including the uncertainties associated with sparse or inadequate data, uncertainties in the models used to calculate flood variables, uncertainties and variabilities in the failure limits of components and structures, uncertain increases in component-failure rates in abnormal flooding environments, and other uncertainties associated with risk estimation (see Chapter 12).

7. There are unique aspects of human interactions that must be taken into account in flood-risk analysis. They include the effects of warning time, if any, to shut the plant down, the conflicts between flood mitigation and plant operations, and the effects of stress.

11.4.3.3 Flooding-Hazard Analysis

The objective of a flooding-hazard analysis is to establish the relationship between the frequency and the magnitude of each flood variable to be analyzed. In the context of the flood-risk equation (11-34), this entails the development of the flood-hazard intensities $h(\underset{\sim}{x})$. Since the approaches to quantification are somewhat specific to the various causes, the methods used for external and internal floods are discussed separately.

External Floods

The first step in analyzing the hazards of external floods is the selection of the causes of flooding and the appropriate flood variables for the site. An exhaustive list of external flood causes is normally found in Chapter 2 of the safety analysis report in the section on hydrology. Depending on the site, these causes may include the following:

1. River flooding.
2. Upstream dam failure.
3. Failure of dikes and levees.
4. Tsunamis.
5. Surges.
6. Seiches.
7. Wind waves.
8. Precipitation.
9. Snow melt.

Item 2, upstream dam failures includes all secondary causes (e.g., earthquakes, overtopping, antecedent dam failures), and item 8, precipitation, includes hurricanes and sequences of storms.

Of the variables associated with a flood that can be related to an assessment of the damage to plant structures, flood height is the most important since little flood-induced damage can be postulated unless the flood height exceeds some minimum level. One possible exception is the blockage of cooling-water intakes with trash, which could result from floods of moderate heights. This minimum level might be the grade elevation of the plant structures, the minimum level at which offsite-power supplies might be damaged, or the minimum level at which a flood-protection system would fail. Certainly, for the latter case the failure of the flood-protection systems should be expressed probabilistically. Hence, a risk quantification of external floods would usually be expected to include an assessment of the hazard curve for flood height.

At least part of the flood-height distribution can be estimated by a statistical analysis of data. In the case of river sites, data are usually analyzed in terms of flow rate instead of height since rarely are data available at the precise location of the reactor site along the river. An example of data on maximum annual flows at a river site is given in Table 11-7 (Wall, 1974). In Figures 11-12 and 11-13, distributions are fit to the data, Figure 11-12 showing the lognormal and log-Pearson type III distributions, and Figure 11-13 showing one of the extreme-value distributions. The differences among the various fits within the range of the data (i.e., for river flows of less than about 50,000 cfs) are negligible in comparison with the magnitude of uncertainties normally encountered in risk assessment. However, when each of these curves is extrapolated for the river flows predicted to exceed the design-basis flood at a frequency of 10^{-4} per year, the following results are obtained (Wall, 1974):

Type of fit	Flow (cfs)	Water elevation (ft)
Lognormal (fitted by eye)	100,000	922
Log-Pearson type III	71,000	918
Extreme value (fitted by maximum-likelihood method)	77,000	919

Table 11-7. Maximum daily discharge at St. Cloud, Minnesota, for water-years 1927 to 1970[a]

Water-year	Flow (cfs)	Water-year	Flow (cfs)
1927	16,885	1949	8,857
1928	13,055	1950	31,920
1929	15,910	1951	19,717
1930	17,076	1952	37,900
1931	10,371	1953	23,110
1932	8,334	1954	20,500
1933	13,409	1955	13,370
1934	5,707	1956	18,460
1935	6,531	1957	19,620
1936	12,139	1958	6,609
1937	13,239	1959	16,375
1938	24,840	1960	14,290
1939	15,598	1961	9,860
1940	14,199	1962	25,500
1941	20,755	1963	12,345
1942	20,835	1964	15,570
1943	27,374	1965	46,780
1944	25,400	1966	26,350
1945	24,216	1967	23,253
1946	26,275	1968	17,746
1947	18,574	1969	39,366
1948	19,286	1970	19,780

[a]From Wall (1980).

Hence, the extrapolation of the fitted curves to a 10^{-4} exceedence frequency results in a difference of as much as 30 percent in the river flow rate, but an increase in water elevation of only 3 feet. Although a difference of 3 feet may be small, such a difference could be important in view of the threshold effect of rising water levels.

The grade elevation for the plant at the reference site is 935 feet, and the water level corresponding to the probable maximum flood was set at 939.2 feet (365,000 cfs). Clearly, the extrapolation of the fitted curves in Figures 11-12 and 11-13 to these extreme flood levels would be excessive. Hence, the curve-fitting techniques alone are in most, if not all, cases insufficient for estimating the hazard frequencies associated with floods exceeding the design-basis floods. The tails of the distribution must be estimated by using some sources of information other than the flood data.

One approach to estimating the tails of the hazard curve is to estimate the total frequency of the sequence of events that is used to define

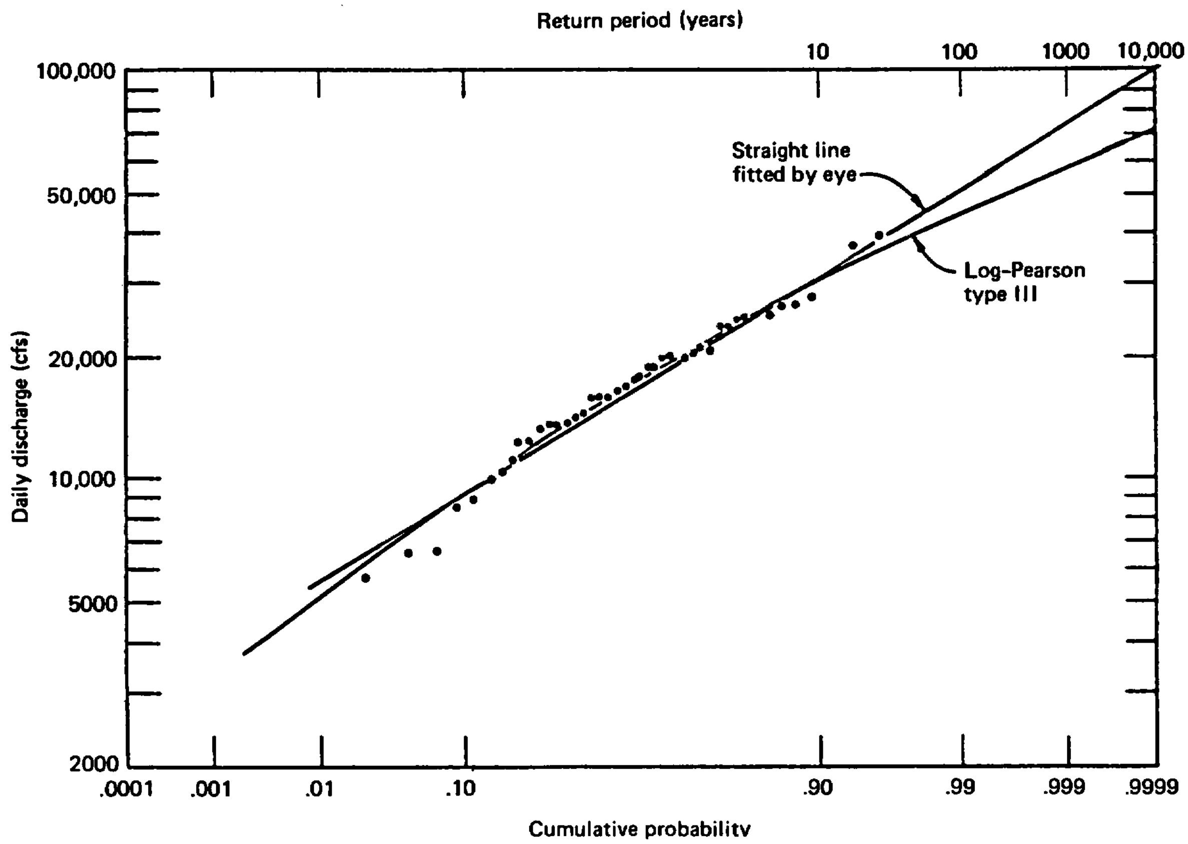


Figure 11-12. Flood data of Table 11-7 plotted on lognormal probability paper. From Wall (1974).

the probable maximum flood (PMF). For the reference site this estimate was made under the assumption of a 100-year snow cover, followed by the maximum historical temperature sequence and the occurrence of the probable maximum precipitation (PMP). The frequency of this sequence could be estimated by

$$\phi(\ell_i^*) = \phi(S)\ \Pr\{T|S\}\ \Pr\{R|S,T\} \tag{11-35}$$

where

$\phi(\ell_i^*)$ = frequency of exceeding the PMF.

$\phi(S)$ = frequency of exceeding the snow cover assumed in the PMF.

$\Pr\{T|S\}$ = conditional probability of a maximum temperature sequence or worse, given the snow cover S.*

$\Pr\{R|S,T\}$ = conditional probability of a PMP, given the sequence S,T.*

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*The temperature and precipitation must occur within a specified period of time to produce the PMF.

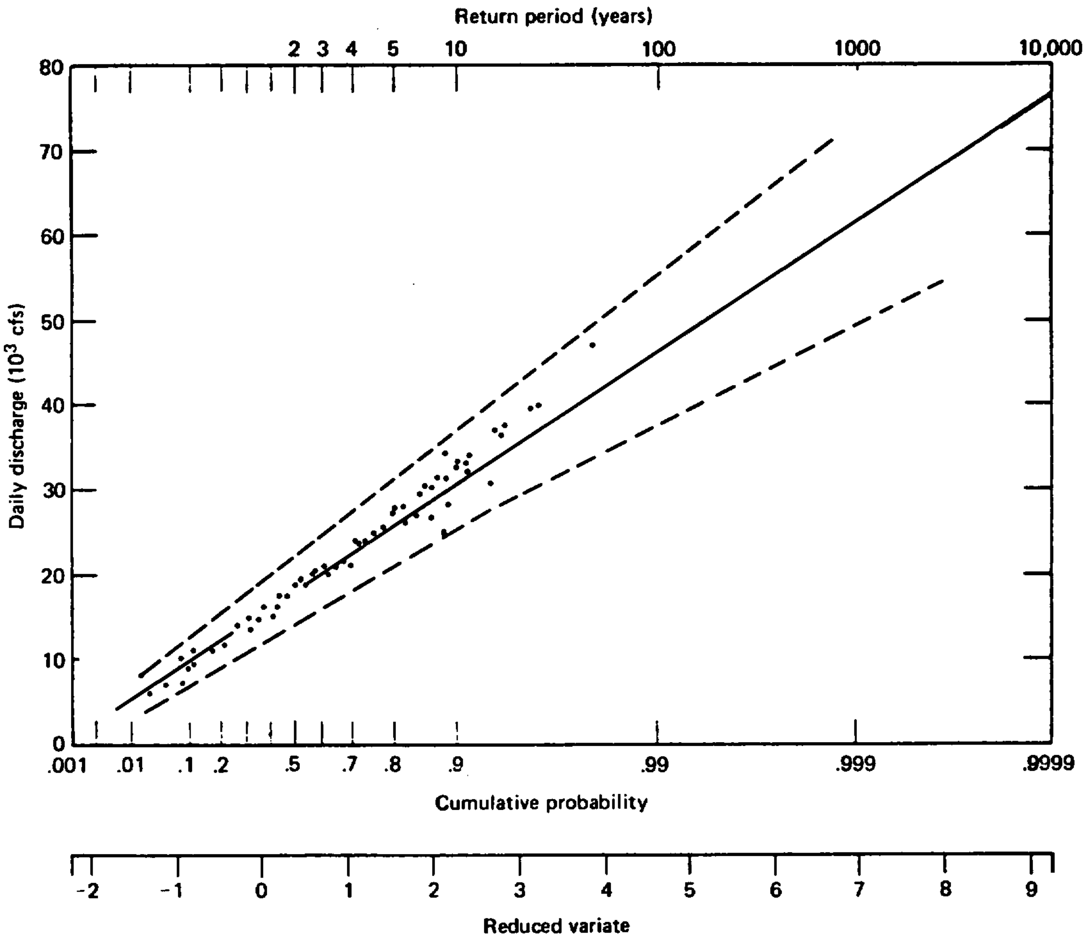

Figure 11-13. Extreme-value graph of flood data of Table 11-7 showing control curves. From Wall (1974).

Data on S, T, and R can be obtained, and the same type of statistical analysis can be performed as described above for river-flow data if it can be assumed that the snow levels, temperature sequence, and precipitation are statistically independent events (see footnote on page 11-80). Statistical independence of events in the sequence causing the flood may not be a good assumption, however, if the sequence includes multiple rainstorms occurring in succession.

Figure 11-14 shows the synthesis from statistical data of a flood-hazard distribution on the flood variable and an estimate of an extreme value on the tail, obtained by using the approach described above. An implied assumption is that there are no sequences of events other than the one used to define the PMF that would produce the flood magnitude ℓ_i^* at frequencies comparable to $\phi(\ell_i^*)$. Since the PMF is defined as the maximum flood resulting from a large number of combinations of candidate event sequences (USNRC, 1977), this is probably a good assumption. The method of synthesizing the two sources of information in Figure 11-14 is simply to draw a smooth curve connecting the fitted curve to the extreme point estimated from Equation 11-35. The indicated probability intervals represent uncertainties in developing the curve.

Sources of uncertainty include those associated with the curve fitting of the data; these can be calculated in terms of confidence limits on the parameters of the fitted line (see Figure 11-13). Similarly, confidence limits can be estimated for each term in Equation 11-35 and appropriately combined by moment-propagation, Monte Carlo, or discrete-probability-arithmetic methods (see Chapter 12) to obtain probability intervals for $\phi(\ell_i^*)$. Other sources of uncertainty that should be incorporated into the above intervals must be estimated subjectively. They include all sources of uncertainty not represented in the data base, such as the possible inapplicability of data because of changes in site hydrology and undefined sequences of natural events that might change the tails of the curve.

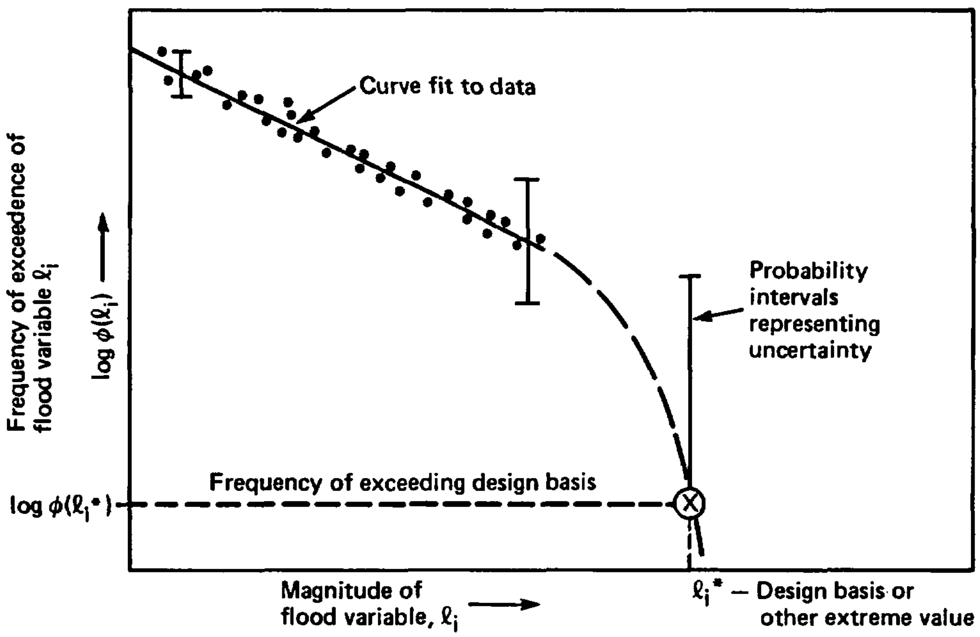

Figure 11-14. Results of a hypothetical hazard analysis of flood variable ℓ_i.

Internal Floods

The analysis of internal flood hazards is conceptually similar to that described in Section 11.3 for the analysis of fire hazards. The hazard analysis in both cases consists of a qualitative phase, in which specific cases are selected for quantification, and a quantitative phase, which provides an estimate of the frequency of exceeding various levels of magnitude. One of the most significant differences between floods and fires from the perspective of how the analysis is carried out is that the specific sources of flooding can be more easily and completely enumerated and floods are very likely to propagate to adjacent compartments, whereas fires are generally confined to rather small areas. These aspects are taken into account in the flood-location screening methods described below.

The identification of important locations must be made from two perspectives. It is necessary to identify both the source locations (i.e., the locations where floods are most likely to start) and the critical impact locations (i.e., the locations where the existence of a flooding condition would have the greatest impact on the availability of key safety-related systems). Both types of location must be considered because of the possibility that the flooding will propagate from one location to another.

One method for ranking locations in terms of the impact of a flooding condition on the risk of reactor accidents is to perform, for the major plant systems, a special type of qualitative fault-tree analysis that takes into account the location of system components. This procedure, illustrated in Figure 11-15, starts by constructing a fault tree for the top event "core melt due to internal flood." The fault tree is developed under the assumption that a postulated flood causes a transient or an initiating event for a

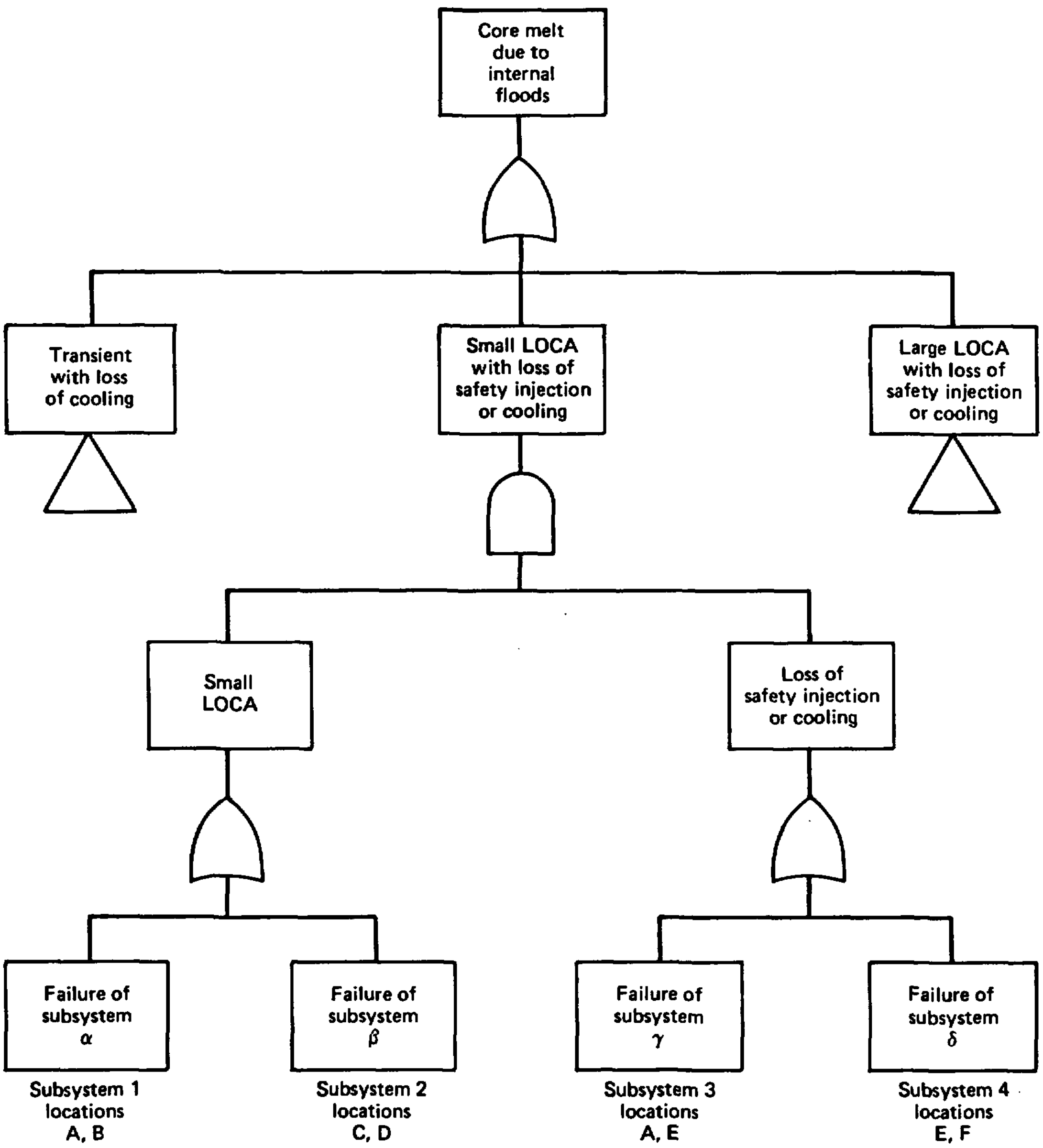

Figure 11-15. Fault tree for identifying important flood-impact locations. The triangles represent events not developed in this example.

small or a large LOCA, or that it causes the failure of a system or subsystem required to mitigate the initiating event, or a combination of these. The tree need be developed only to the level of major subsystems (e.g., high-pressure-injection train A) so that the singular effect of the loss of groups of components in specific locations can be resolved. It is necessary, however, to include support systems whose failure has a significant impact on the availability of the main safety-related systems.

The example of Figure 11-15 is a simplified illustration: the subtree for a small LOCA is developed to identify four subsystems denoted α, β, γ, and δ. In a practical application of this method, the tree would be much larger, and the subsystems would typically consist of redundant trains of components; hence, there would be more AND gates near the bottom of the tree. After the tree has been developed to the subsystem level, the locations of all components in each subsystem are itemized. In this simple example, the four subsystems are seen to have components in a total of six locations labeled A through F.

The next step in this approach is to determine the minimal cut sets of the fault tree, which in this case are

$$(\alpha,\gamma);\ (\alpha,\delta);\ (\beta,\gamma);\ (\beta,\delta)$$

The relative importance of each location is determined by postulating that a flooding condition exists in each location, one at a time, and assuming that all the subsystems in that location are failed with a probability of unity. Conditional minimal cut sets for each flood location are then determined in two steps: (1) by modifying each of the original cut sets to remove the subsystems associated with each location and (2) by reducing the remaining cut sets to minimal cut sets by eliminating the supersets. The set (β,γ) is a superset of (γ), for example. The cut-set analysis for this example is presented in Table 11-8.

Table 11-8. Fault-tree quantification to determine the impact importance of flood locations[a]

Subsystem-level cut sets	Conditional cut sets given flood-induced failure at location j					
	A	B	C	D	E	F
(α,γ)	(1)	(δ)	(α,δ)	(α,δ)	(α)	(α,δ)
(α,δ)	(δ)	(δ)	(α,δ)	(α,δ)	(α)	(α)
(β,γ)	(β)	(β,δ)	(δ)	(δ)	(β)	(β,δ)
(β,δ)	(β,δ)	(β,δ)	(δ)	(δ)	(β)	(β)
Conditional minimal cut sets	(1)	(γ), (δ)	(γ), (δ)	(γ), (δ)	(α), (β)	(α), (β)
Conditional top-event frequency	1	10^{-2}	10^{-2}	10^{-2}	10^{-1}	10^{-1}

[a]See Figure 11-15.

The simplest way to rank locations is by the size of the minimal cut sets remaining after postulating the flood. In this case location A would simply rank above all the rest since the location failure itself produces the top event. The remaining locations cannot be distinguished since each results in two single-event minimal cut sets. Note that the correlation between cut-set size and the frequency of failure is only approximate.

If system unavailabilities from causes independent of the flood are known, a more effective ranking can be made by estimating the conditional frequency of core melt given each failed location. In the example, the following subsystem unavailabilities are assumed: 10^{-1} for α, 10^{-3} for β, 10^{-2} for γ, and 10^{-4} for δ. The quantification of the fault tree gives greater resolution in ranking the impact locations, as would be expected. The quantitative approach results in the following importance ranking from most to least important:

1. A
2. E, F
3. B, C, D

Note that, if a bounding estimate can be obtained for the frequency of a flooding condition in each of the locations, a bound on the risk due to flooding can be obtained at this point. Such an estimate would conservatively neglect fragility--that is, the conditional frequency of failure given a flooding condition in each location. Several computer codes are available to aid in the qualitative analysis of locations in fault trees. Discussed in Sections 3.7 and 6.6, these codes include COMCAN, BACFIRE, and WAMCOM.

After the important impact locations have been determined, it is necessary to evaluate the source locations where floods can start. The analyst starts by listing the major sources of water at the plant, including the major tanks and systems that supply, circulate, and process water. Such systems would include, for example, the circulating-water, condensate, feedwater, service-water, component-cooling water, makeup and purification, spent-fuel pool, reactor-coolant, safety injection, and decay-heat-removal systems. A qualitative evaluation should be performed on each system and flooding source to identify and select those for quantification.

One useful technique to aid in the selection of source locations is a failure modes and effects analysis (FMEA) structured especially for this application. A similar approach has been successfully applied to the evaluation of important fire locations. An FMEA format specialized for floods is shown in Figure 11-16. The source locations found to have the relatively greatest potential for propagating to one or more important impact locations are selected for quantitative analysis.

The hazard analysis for internal flooding is completed by estimating the frequency of flood initiation, at each source location selected for analysis by the FMEA procedure, as a function of flood severity. A particular initiating flood might produce various degrees of flooding, depending on, for example, the timing and the success or failure of various mitigating actions, such as the shutting down of pumps or the closing of isolation

System or flood source	Subsystem or component	Location	Flood rate range (gpm)	Inventory of maximum estimated flood quantity (gal)	Colocated components, systems	Flood barriers and mitigating actions	Flood pathways to important impact locations
Circulating-water system	Expansion joint	Near condenser at el. 105 ft in turbine building	10,000	300,000 if terminated in 30 min	Condensate pumps, condensate booster pumps		

Figure 11-16. Example of FMEA format for evaluating flood-source locations.

valves. A distribution of flood magnitudes at a given location can be developed either directly from the data base or with the aid of a specialized event tree that is conceptually illustrated in Figure 11-17.

There is a substantial body of statistical data that is applicable to the hazard analysis of internal floods. Nuclear Power Experience (Verna, 1981) lists about 60 incidents at U.S. nuclear power plants that involved

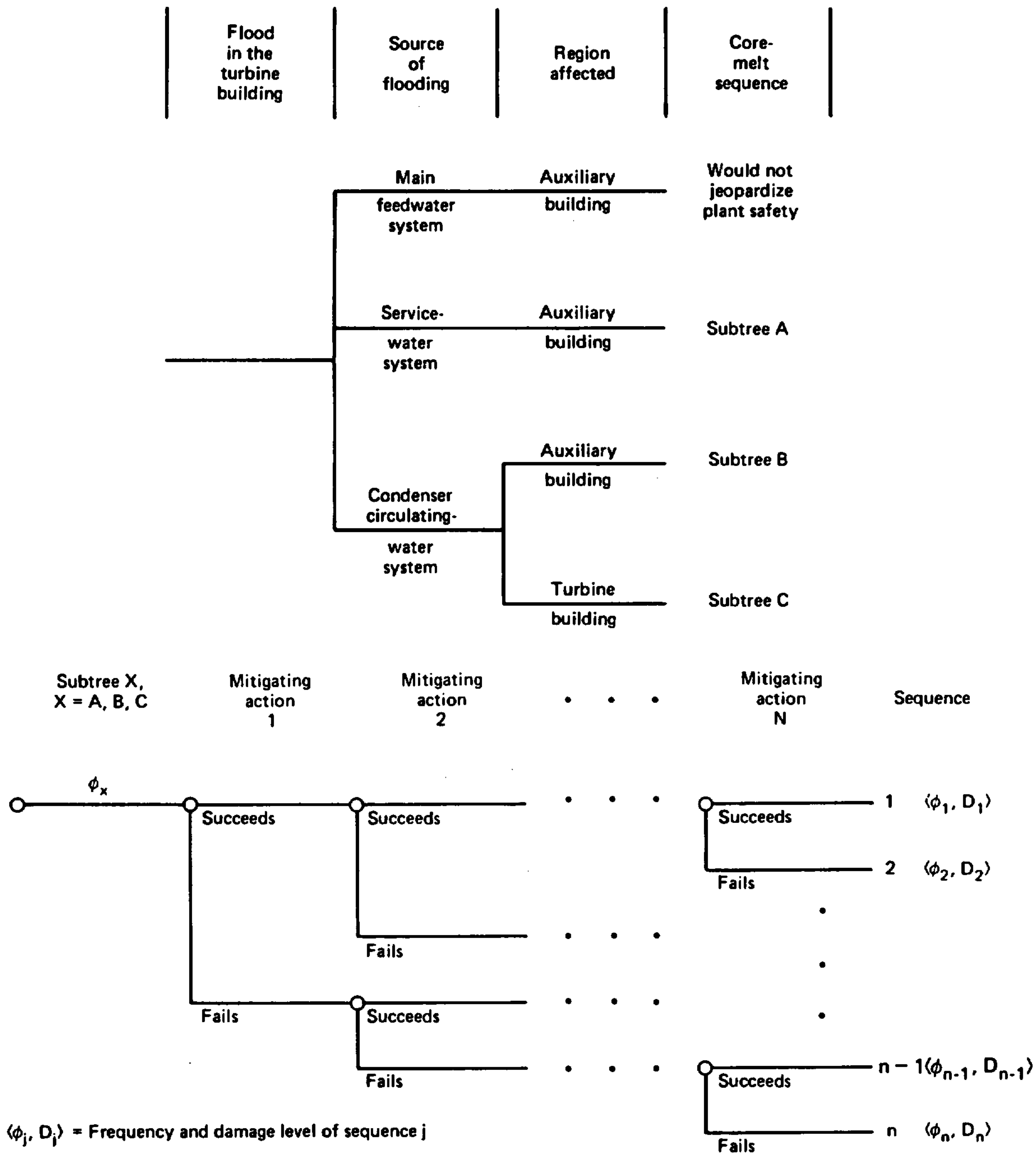

Figure 11-17. Event tree for developing frequency-magnitude estimates for the hazards of internal floods.

Table 11-9. Flooding frequencies for turbine and auxiliary buildings

Location	Severity level	Flooding frequency (per reactor-year) 5th percentile	Median	95th percentile	Mean
Auxiliary building	Small	2.0×10^{-6}	3.4×10^{-4}	1.0×10^{-2}	3.1×10^{-3}
	Moderate	1.6×10^{-4}	7.4×10^{-3}	3.1×10^{-2}	1.5×10^{-2}
	Large	1.0×10^{-6}	4.5×10^{-4}	2.0×10^{-2}	6.3×10^{-3}
	Moderate and large	2.5×10^{-5}	3.3×10^{-3}	1.6×10^{-2}	1.6×10^{-2}
Turbine building					
Service-water source	Moderate to large	2.9×10^{-7}	1.4×10^{-3}	2.5×10^{-2}	4.9×10^{-3}
Circulating-water source	Moderate to large	2.2×10^{-3}	1.2×10^{-2}	1.3×10^{-1}	2.8×10^{-2}

flooding of some sort. Other data of interest are the component-failure and pipe-failure data that have been analyzed and used in the assessment of loss-of-coolant accidents (see Chapter 5). These data can be analyzed to provide an estimate of the frequency of floods as a function of magnitude at specific locations. Shown in Table 11-9 are the results for floods in the auxiliary building and the turbine building. The following definitions of flood severity were used to categorize the data:

1. Small. Flooding on the order of hundreds of gallons (e.g., valve pit flooding, flooding of an instrument, or flooding within a component).

2. Moderate. Flooding on the order of several thousands of gallons (e.g., a few feet of water on the floor of a typical pump room).

3. Large. Flooding on the order of tens of thousands of gallons (e.g., a few feet of water in large rooms, very deep water [more than 10 feet] in a typical pump room).

4. Very large. Flooding on the order of hundreds of thousands of gallons (e.g., floods involving circulating-water or service-water piping).

A Bayesian procedure, identical with that described in Chapter 5 for the analysis of component and initiating-event data, can be used as indicated in Table 11-9 to express the level of uncertainty in the frequency estimates. As indicated in Section 11.4.3.3 for external floods, the quantification of hazard curves for internal floods must include an assessment of uncertainty by providing a family of curves, each of which is assigned a probability describing the level of uncertainty or state of knowledge.

Special attention should be given to component failures and whether the flooding entails sprays of water or just a rising pool. For example, spurious actuations are possible if an electrical cabinet is sprayed with water. Otherwise, the electrical circuits within the cabinets will be deenergized. Furthermore, special attention should be given to flood termination possibilities (in the case of most internal floods) and recovery of failed systems via local manual actuations. The latter entails a human-error analysis, which is discussed in Chapter 4.

11.4.3.4 Fragility Evaluation

The objective of fragility evaluation is to estimate the frequency of producing each of a number of flood-damage states E_ℓ as a function of the flood intensity $\underset{\sim}{x}$ and response $\underset{\sim}{y}$, denoted by $f_{e,\ell}(\underset{\sim}{y})$. For convenience and simplicity, it is assumed that the continuum of flood-damage states can be adequately approximated by a finite number of discrete states. This does not differ from the practice of using a finite number of release categories to approximate the continuum of possible release magnitudes that could result from a reactor accident.

The first step in fragility evaluation is to define the flood-damage states for analysis. These could be expressed as specific combinations of structural failures that might result from external floods or the occurrence of flooding at various combinations of important impact locations determined from the hazard analysis of internal floods. In the latter case the natural combinations of impact locations would be adjacent locations or those that would most likely be linked by propagation pathways emanating from the flood source. The flood-damage states used by Wagner et al. (1980) consisted of submergence to different flood heights. As noted earlier, however, special attention should be given to sprays of water. This is because under a spray condition the failure mode of some components, such as electrical cabinets, could be quite different from their failure mode under a rising-pool condition.

Although there have been no known attempts to do so, the method described in Section 11.2 for estimating the fragility of structures in terms of safety factors incorporated into the design should be applicable to floods as well. Confidence that this is indeed true is supported by the observation that a similar method has been successfully used in the Indian Point PRA (PASNY, 1982) to estimate the fragility of structures subjected to extreme winds and wind-generated missiles. The major difference in applying this technique to floods is that the calculation of structural loads and integrity must account for failure mechanisms unique to floods, which include wave runup and impact forces, erosion, missile strikes, liquefaction, ponding, overturning, sliding, hydrostatic loading, and leakage. Another failure mechanism is the blockage of cooling-water intakes by trash, which can occur in floods less severe than the probable maximum flood.

In addition to the fragility of structures, it is necessary to consider, for those event sequences in which the pertinent structures do not fail, the fragility of components inside the critical impact locations identified in the hazard analysis. A conservative approach is to assume that the components inside a room are failed if a flooding condition propagates to that room or location. The term "flooding condition," as used here, includes full or partial submergence, spraying, seepage into, or the wetting by any other mode of equipment anywhere inside the room. If this conservative approach is taken for internal floods, the entire fragility evaluation can be incorporated into the event tree of Figure 11-17. A less conservative approach was that taken by Wagner et al. (1980), who assumed that components fail only when submerged.

11.4.3.5 Plant and System Analysis

The objective of the plant and system analysis is to estimate the frequency of core-damage or core-melt sequences initiated at each of the flood-damage states defined in the flood-hazard analysis. This phase of the flooding-risk analysis uses the basic event- and fault-tree method described in Chapter 3. It is important, however, to ensure that this basic method of analysis accounts for the boundary conditions associated with each flood-damage state.

There are several different approaches to the plant and system analysis, each of which uses event trees, fault trees, or both. Variations on these approaches are described in Section 10.3.6.

11.4.3.6 Release-Frequency Analysis

The objective of release-frequency analysis is to estimate the conditional frequency of exceeding levels of accident consequences, given the occurrence of each flood-damage state. In the notation of Equation 11-34, this quantity is $f_{k|S,j}(z)$. The methods described in Chapters 7 and 9 for analyzing the containment event tree and the consequences of core-melt accidents should be fully applicable to flood-induced accidents, with the exception that dependences between the cause of the flood and certain factors that might affect offsite consequences must be taken into account. These dependences include weather conditions, the effects of the flood on emergency plans and evacuation, and liquid pathways for radionuclides. In the case of internal floods, there is no need to carry out a special analysis of release frequencies because these dependences would not apply.

11.4.4 INFORMATION REQUIREMENTS

The information required to perform a risk analysis of flooding consists of the following:

1. Description of plant systems, including the location of components and systems within structures; a set of general arrangement, structural, piping, electrical, and equipment drawings.

2. Safety analysis report, especially the chapters on the hydrologic characteristics of the site, the meteorological and topographic

features of the region, design criteria, applicable codes and standards.

3. Reports on qualification and preservice tests as well as inservice inspections.

4. Results of the plant and systems analysis for internal accident initiators, including data on the component-failure rates.

5. Statistical data from the National Weather Service, U.S. Army Corps of Engineers, and other sources on precipitation conditions contributing to floods. A summary of these data requirements for various types of sites is presented in Table 11-10.

6. A compilation of licensee event reports involving flooding at nuclear power plants like that provided by Verna (1982).

Table 11-10. Statistical data requirements for the analysis of external floods at various types of site

	Applicability of data type to site			
Data-source description	River site	Atlantic Coast or Gulf of Mexico site	Great Lakes site	West Coast site
Local historical point-precipitation information	X	X	X	X
Tropical storm history	X	X	X	X
Storm-surge history and potential		X	X	X
Seiche history and potential			X	
Characteristics of historical river floods	X			
History of astronomical tides (including initial rise effects)		X		X
Snow pack and melt characteristics	X		X	
Wind-wave and wave-setup potential	X	X	X	X
Basic hydrosphere characteristics	X	X	X	X
Historical geoseismic activity and potential				X
Locations and characteristics of dams, levees, etc.	X	X	X	X

11.4.5 PROCEDURE

The methods described in Section 11.4.3 are summarized below in the form of a step-by-step procedure for analyzing the risk from flooding.

Task 1: Flood-Hazard Analysis

1. Identify external and internal causes of flooding.

2. Identify important flood variables and failure mechanisms.

3. Determine the critical flood-impact locations.

4. Estimate the frequency-of-exceedence curve for each flood variable.

5. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

Task 2: Fragility and Vulnerability Evaluation

1. Define flood-damage states.

2. Determine the susceptibility of components and structures to each flood-failure mechanism.

3. Identify actions to mitigate flood damage.

4. Estimate the frequency of each flood-damage state as a function of flood magnitude.

5. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

Task 3: Plant and System Analysis

1. Develop the event- or fault-tree logic that defines the flood-damage states and relates them to core-damage states.

2. Apply component- and system-failure boundary conditions to the event- and fault-tree logic.

3. Estimate the frequency of core-damage accidents initiated by each flood-damage state.

4. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

Task 4: Release-Frequency Analysis

1. Identify dependences between floods, meteorological conditions, and evacuation procedures.

2. Develop data for liquid-pathway modeling.

3. Estimate the frequency-of-exceedence curve for the consequences resulting from each flood-damage state.

4. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

11.5 ASSURANCE OF TECHNICAL QUALITY

The provisions described in Chapter 2 for the assurance of technical quality are applicable to a seismic risk analysis as well. The sources of data, design reports, and material and qualification test results should be documented. Since the risk analysis is based heavily on assumptions and engineering judgment, it is essential to have all aspects of the analysis thoroughly reviewed by peers.

NOMENCLATURE

A	seismic capacity of a component expressed in terms of ground acceleration; a random variable
$\check{A}$	median ground-acceleration capacity
A_{rms}	root-mean-square acceleration
A_{SSE}	local response parameter specified for the reference earthquake (e.g., safe-shutdown earthquake)
a	specific value of the random variable A
a_0	a parameter in the recurrence relationship by Gutenberg and Richter (1942)
a_D	effective peak ground acceleration
a_{pi}	instrumental peak ground acceleration
a_s	sustained maximum ground acceleration
b_0	Gutenberg-Richter slope parameter
b_1, b_2, b_3	constants in the attenuation equation
C	seismic capacity of the component
F	factor of safety
$\check{F}$	median factor of safety
F_C	capacity factor of safety
F_{RE}	factor of safety in equipment-response computations
F_{RS}	factor of safety in the structural response analysis
F_S	strength factor
F	inelastic-energy-absorption factor
f'	frequency of component failure at nonexceedence-probability level of Q
$F_{A\|m_i,r_j}(a)$	frequency with which the ground-motion parameter A exceeds a value a given an earthquake of magnitude m_i at a distance r_j
f_c	core-melt frequency
$f_\ell(m_i)$	conditional frequency with which an earthquake at the source has a magnitude equal to m_i

$f_{\ell}(r_j)$	frequency with which the source-to-site distance is r_j given an earthquake on the ℓ^{th} source
$f_s(a)$	conditional frequency of plant-system failure leading to core melt for an effective peak ground acceleration equal to a
$H(a)$	cumulative annual frequency of occurrence of earthquakes that cause ground-motion parameter values less than or equal to a
$h(a)$	annual frequency of earthquakes with ground-motion parameter values between a and a + Δa
I_0	epicentral intensity (MM) of the earthquake site intensity
I_s	site intensity
K_p	A function of acceptable frequency p, relating the value of a_D to A_{rms}
ℓ	a seismic source
M_s, M_1, M_2, M_3	system-failure events
m	Richter (local) magnitude; m_i a specific value
m_0	earthquake magnitude below which damage rarely occurs
m_b	bodywave magnitude, can be related to m
m_m	upper-bound magnitude for the source
N	number of earthquakes per year exceeding magnitude m_i
P, Q	nonexceedence probability
P_N	normal operating load or stress
P_T	total load (stress) on the components (i.e., the sum of the seismic load and the normal operating load)
R	distance of the earthquake energy center (epicenter) from the site, a random variable
r_j	a specific value of R
β_0	$b_0 \ln 10$, where b_0 is the Gutenberg-Richter slope parameter in the recurrence equation
$\beta_{(.),R}$	logarithmic standard deviation of the variable reflecting inherent randomness
$\beta_{(.),U}$	logarithmic standard deviation of the variable reflecting uncertainty

$\beta_{(.)}$	logarithmic standard deviation representing total variability
$\varepsilon_{(.),R}$	random variable (with unit median) representing the inherent randomness in the variable designated in the parentheses
$\varepsilon_{(.),U}$	random variable (with unit median) representing the uncertainty in the median value of the variable (.)
μ	allowable ductility ratio
ν_{ℓ}	mean number of earthquakes per year on the ℓ^{th} source; activity rate of the source
$\nu_{(a)}$	total mean number of earthquakes per year in which A exceeds a at the site
$\nu_{\ell}(a)$	mean number of earthquake events per year in which A exceeds a at the site because of an earthquake on the ℓ^{th} seismic source
$\Phi(.)$	standard Gaussian cumulative distribution function
$\vee$	OR symbol
$\wedge$	AND symbol

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